E. M. Dean

ANL-7665

Argonne National Laboratory

A STUDY OF THE RESPONSE OF THE EBR-II PLANT PROTECTIVE SYSTEM TO HYPOTHETICAL MALFUNCTIONS IN THE REACTOR SYSTEM

by

A. V. Campise

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.65

ANL-7665
Reactor Technology

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

A STUDY OF THE RESPONSE OF THE EBR-II PLANT PROTECTIVE SYSTEM TO HYPOTHETICAL MALFUNCTIONS IN THE REACTOR SYSTEM

by

A. V. Campise

EBR-II Project

June 1970

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

A STUDY OF THE RESPONSE OF THE EBR-II PLANT PROTECTIVE SYSTEM TO HYPOTHETICAL MALFUNCTIONS IN THE REACTOR SYSTEM

by

A. V. Campise

ABSTRACT

The prompt response of the EBR-II plant protective system to hypothetical malfunctions of reactor components has been investigated by means of a comprehensive dynamic simulation and analysis. The hypothetical malfunctions investigated are those listed in the EBR-II Hazard Summary Report. The AIROS-IIA dynamic-simulation digital code was used to model the principal neutronic, thermal, and hydraulic characteristics of typical driver-fuel and experimental-oxide-fuel elements. Response times of the EBR-II protective system were taken from in situ measurements made at the reactor plant on circuits sensing neutron flux, coolant flow rates, and temperatures. The studies indicate that the present EBR-II plant protective system provides responsive and redundant protection for all hypothetical malfunctions of components listed in the Hazard Summary Report.

I. INTRODUCTION AND OBJECTIVES

Plant safety is of primary importance in the successful operation of EBR-II as an irradiation facility.* The assurance of safety has required a continuing surveillance of the dynamic characteristics of all systems in the reactor plant. This surveillance has included studies of the reliability, redundancy, and independent performance of all EBR-II plant control and protective systems under emergency operating conditions.

The studies reported here were directed specifically at understanding the response of the current EBR-II plant protective system (PPS) to the various malfunctions of components described in the EBR-II Hazard Summary Report (HSR) and Addendum.[1,2] Commensurate with the assumptions in the HSR, various portions of the PPS were assumed to be inoperative during various plant operations. Updated studies in this area were needed,

*Experimental Breeder Reactor II (EBR-II) is the United States Atomic Energy Commission's primary facility for irradiation tests of fuels and materials for the Liquid Metal Fast Breeder Reactor Program.

because of design improvements and programmatic changes incorporated into the EBR-II plant. The expanded role of EBR-II as an irradiation facility had brought into importance the relationship between EBR-II driver fuel and experimental fuels. Proposed improvements in EBR-II driver fuel and proposed operation of EBR-II at 62.5 MWt (rather than 50 MWt) also came into consideration.

The basic objective of the present report has been to formulate a firm technical basis for all PPS setpoints and instrument responses in circuits involving reactor period, power level, coolant flow, and temperature. This formulation is based on maintaining transient temperatures of all materials within a safe range during occurrence of the malfunctions described in the HSR. The aim has not been to consider in detail the electronic circuits of the PPS, but to analyze their actions in relation to malfunctions in the reactor.

Basic questions must be answered in reviewing any PPS response to an operational abnormality. These questions are:

1. Why is protection needed?
2. Against what do we protect?
3. At what level is protection required?
4. What uncertainties must be considered?
5. How fast must the protective system respond?

In subsequent discussions, many terms will be used that may require clarification. These terms are defined below.

A. Definitions

1. Reactor operating parameters: The core and blanket parameters during power operations (i.e., reactor period, power level, coolant flow, and material temperatures).

2. Restraint: An upper boundary value for a particular reactor operating parameter.

3. Protective restraint: The upper boundary value below which the PPS must confine a particular reactor operating parameter by its prompt, responsive action.

4. Transient material performance restraint: A value at which an undesirable phase change will occur in a particular reactor material (i.e., sodium boiling, fuel melting, or uranium-stainless steel eutectic formation).

5. Protective margin: The margin including all postulated uncertainties between the listed protective restraint and the transient material-performance restraint. (This margin is based on a 3σ confidence limit.)

6. Setpoint: The parameter value at which a signal is released to the control and/or safety rods to terminate reactor operation.

7. Response time: The time interval between release of a trip signal and initial movement of the control and/or safety rods.

8. PPS operating range: The area bounded by setpoint, response time, and protective restraint in which a particular reactor operating parameter is confined by action of the PPS.

B. Approach of Study

The approach taken in this study has been to answer the previously listed questions in the following way:

1. Why is protection needed?

Protection is needed to prevent the basic reactor operating parameters--neutron flux, coolant flow, and temperature of materials--from exceeding acceptable restraints.

2. Against what do we protect?

We protect against all malfunctions of reactor components that will cause the reactor operating parameters to exceed acceptable restraints.

3. At what level is protection required?

Protection is required at parameter levels that ensure confinement of the flux, flow, and temperature values within the protective restraint, that is, the acceptable restraint with allowances for uncertainties.

4. What uncertainties must be considered?

All uncertainties in reactor environment, fabrication of components, physical properties of materials, effects of irradiation, and location of sensors must be considered.

5. How fast must the PPS respond?

The PPS must respond fast enough to maintain the reactor operating parameters within the acceptable restraints with allowances for all known and postulated uncertainties.

C. Aim of Report

The aim of this report, therefore, has been to define an "operating range" for the EBR-II PPS that will ensure that reactor operating parameters do not exceed acceptable restraints, but are confined by the response and action of the PPS.

II. NUCLEAR, FLOW, AND TEMPERATURE PORTIONS OF EBR-II PPS

This section briefly describes some of the instrumentation in the EBR-II PPS. It is followed by a section on the restraints established for transient temperatures of reactor materials; these two sections lead to a discussion on the dynamic simulation of the response of the PPS to malfunctions in the reactor.

The critical reactor parameters studied in this dynamic analysis are flux, flow, and temperature. The portions of the PPS that sense reactor period, power level, temperature, and changes in flow have been studied to identify protective restraints consistent with transient material-performance restraints.

A. Nuclear Instrumentation

The basic nuclear instrumentation in EBR-II consists of 10 separate sensing channels. These channels provide electrical signals to the control system for indication of neutron-flux levels and reactor periods from source level through the full power range of the reactor. The nuclear channels initiate automatic trips of all the control rods during reactor operation, or an automatic trip of the two safety rods during startup or fuel-handling incidents.

Figure 1 is a general layout of the EBR-II reactor showing the location of the major components. In the outer neutron shield are the instrument thimbles, which contain the neutron detectors.

Figure 2 shows the arrangement of the outer neutron shield, which surrounds the reactor vessel. Depicted in the figure is a typical nuclear-instrument thimble (denoted as J4 instrument thimble). Eight nuclear-instrument thimbles positioned in different geometries and locations in the reactor vessel house all the neutron detectors.

The importance and operating range of the 11 nuclear channels are briefly described below.

1. Channels 1-3

Channels 1-3 are log-count-rate channels using fission-counter detectors. They measure neutron flux level and reactor period from source level to approximately 500 W. These channels provide a startup flux-level interlock, which withholds control power until they measure a neutron count rate in excess of 5 counts/sec. They provide a low-power reactor-period trip (25 ± 3 sec) for reactor operation and fuel handling. Power-level trips (1500 ± 500 counts/sec) are provided for fuel handling only.

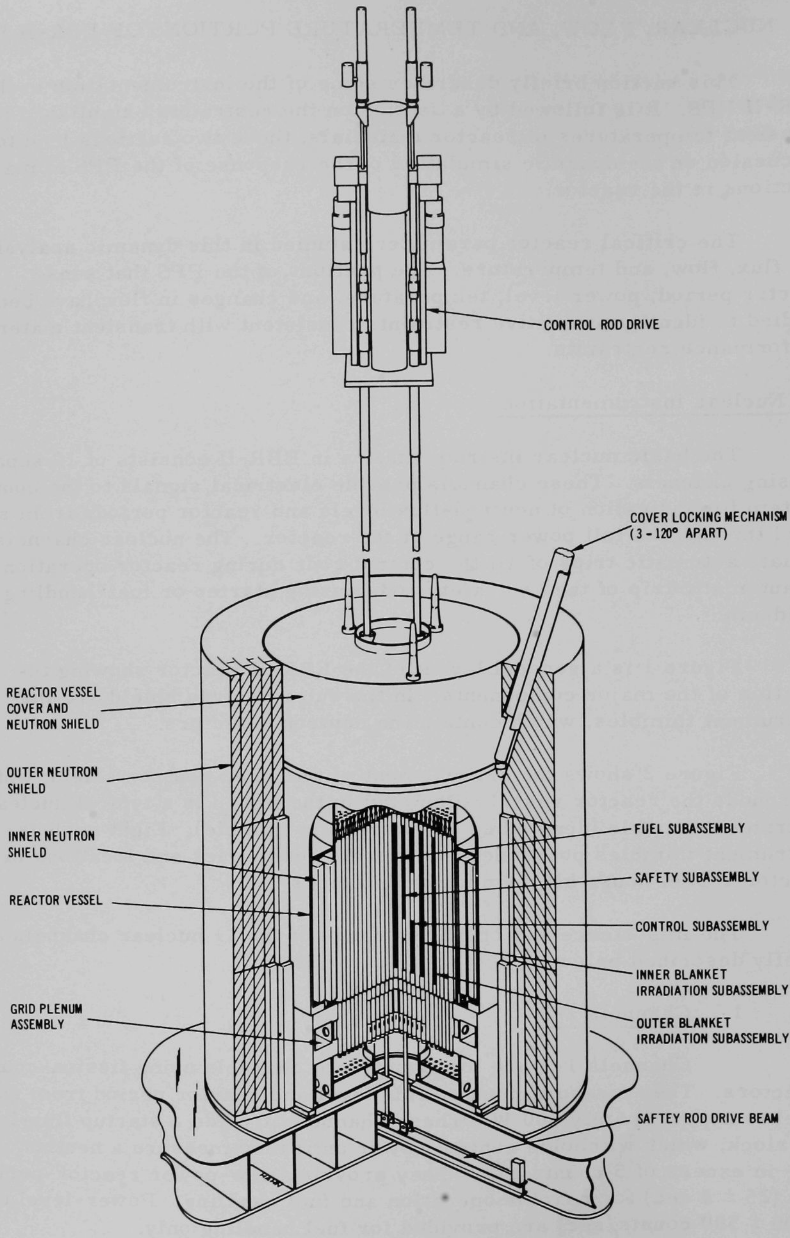


Fig. 1. EBR-II Reactor, Showing Location of Major Components

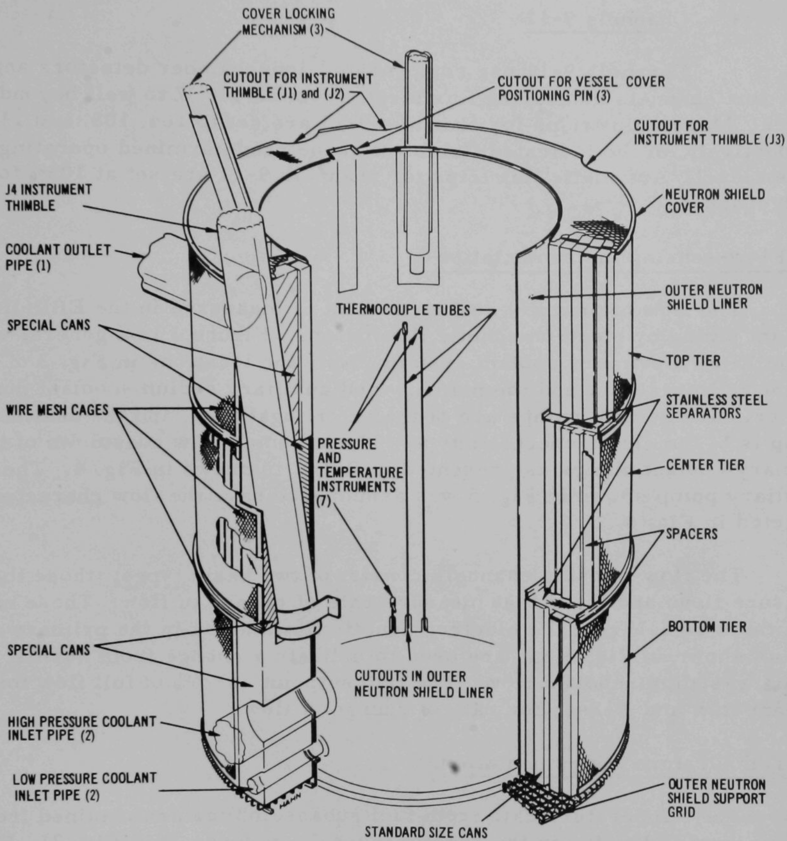


Fig. 2. Arrangement of Outer Neutron Shield around Reactor-vessel Assembly

2. Channels 4-6

Channels 4-6 are log-N power channels using compensated-ion-chamber detectors. They provide reactor-period trips (25 ± 3 sec) and reactor log-N (flux level) from about 100 W to well beyond full reactor power. During reactor startup when the log-N channels reach approximately 6 x 10^{-10} A, the period and high-voltage trips in the log-count-rate channels are automatically bypassed. The reverse is true during reactor shutdown.

3. Channels 7 and 7A

Channels 7 and 7A are linear channels using compensated-ion-chamber detectors. They measure flux level from about 50 W to well beyond full power. Channel 7 is normally an operating channel, with channel 7A available as a backup.

4. Channels 9-11

Channels 9-11 use compensated-ion-chamber detectors and are high-flux channels measuring flux levels from 20,000 W to well beyond full power. Manual flux trips for channels 9-11 are set to 104, 108, and 110%, respectively, of the indicated flux level at the predetermined operating power level. Automatic flux trips for channels 9-11 are set at 105% for each channel.

B. Flow-sensing Instrumentation

The flow of primary sodium coolant is measured in the EBR-II coolant piping by electromagnetic flowmeters. Figure 3 is a general view of the EBR-II primary coolant system. Note the locations in Fig. 3 of the magnetic flowmeters and the primary and auxiliary sodium-coolant pumps. The primary coolant pumps are of the centrifugal type, and the auxiliary pump is of the electromagnetic type. The assumed flow coastdown of the primary coolant pumps as presented in Ref. 2 is shown in Fig. 4. The auxiliary pump shown in Fig. 5 was assumed to have the flow characteristics depicted in Fig. 6.

The flow-sensing channels consist of two basic types: those that measure flow, and those that measure rate of change of flow. These channels take their signal from the electromagnetic flowmeters in the primary system as shown in Fig. 3 and are used to indicate a change from normal in primary-sodium-coolant flow. The trip setpoint is 96% of full flow for low coolant flow and 3%/sec for rate of change of flow.

C. Temperature Instrumentation

The temperature data from fuel subassemblies are obtained from thermocouples located in the reactor-vessel top cover (see Fig. 7). Figure 8 shows the location of these thermocouples with respect to subassembly positions. Figure 9 is an overall view of the top cover, thermocouple locations, and fuel subassemblies.

Two instrument columns carry the thermocouple leads into the reactor vessel. These thermocouple leads are routed to various portions of the reactor core. These columns will be referred to here as the north and south columns. The thermocouples measure representative mixed-mean sodium-outlet temperatures from selected fuel subassemblies in the core.

D. Assumed Initial Conditions

In studying the response and subsequent action of the EBR-II PPS, we assumed three different combinations of initial operating conditions. These

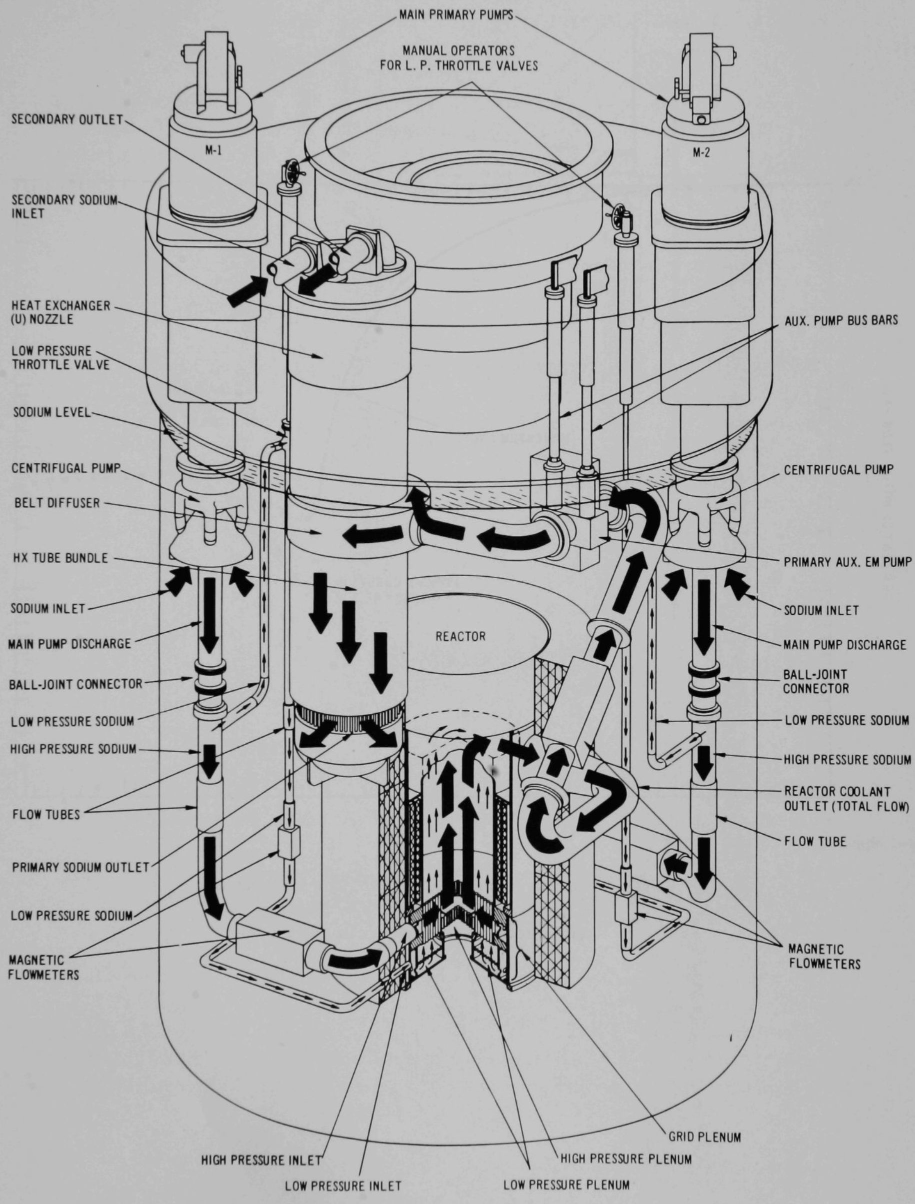


Fig. 3. Primary Coolant System

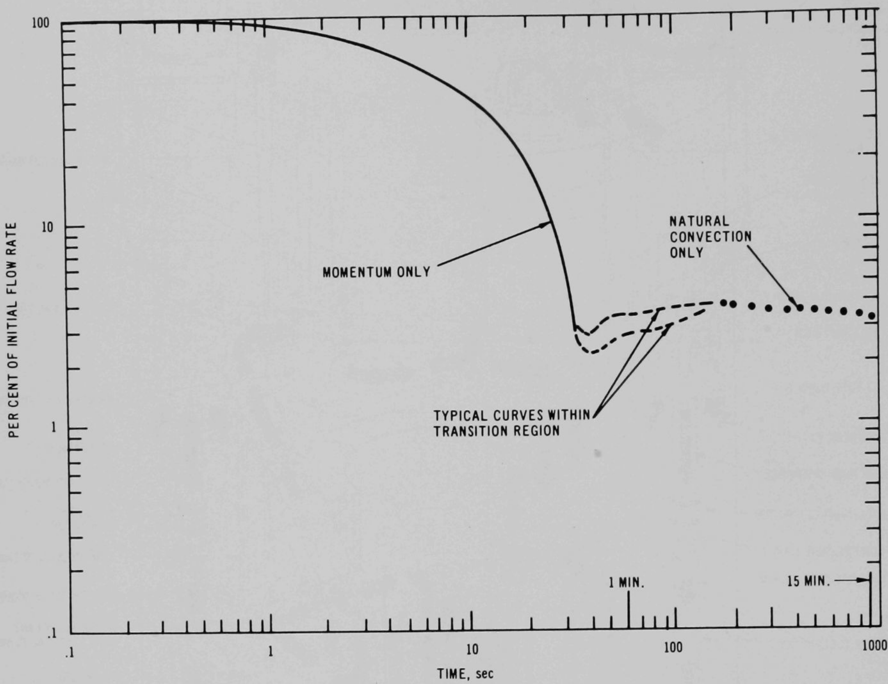


Fig. 4. Primary-system Coolant Flow Rate vs Time after Cessation of All Pumping Power (based on 100% initial reactor power and coolant flow rate)

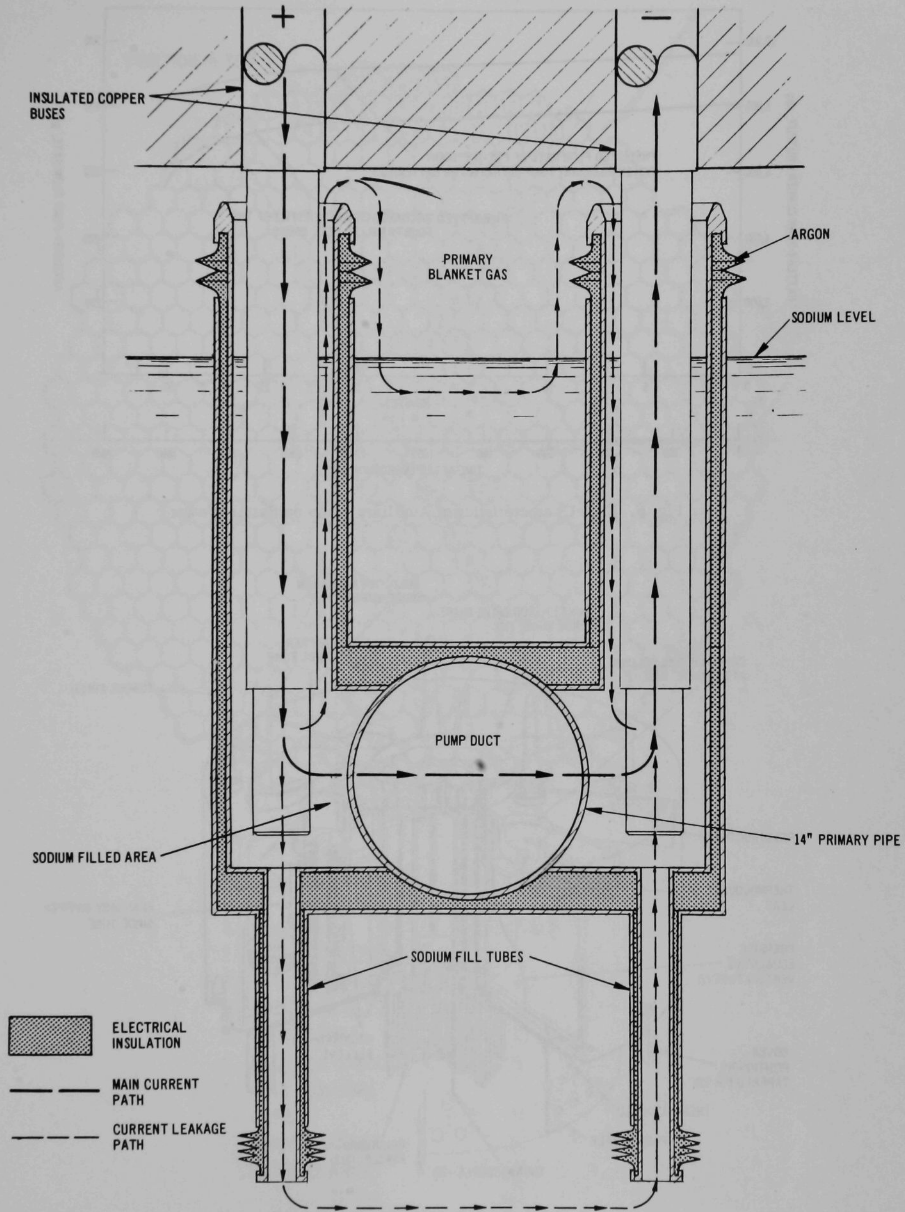


Fig. 5. EBR-II Primary-system Auxiliary Pump

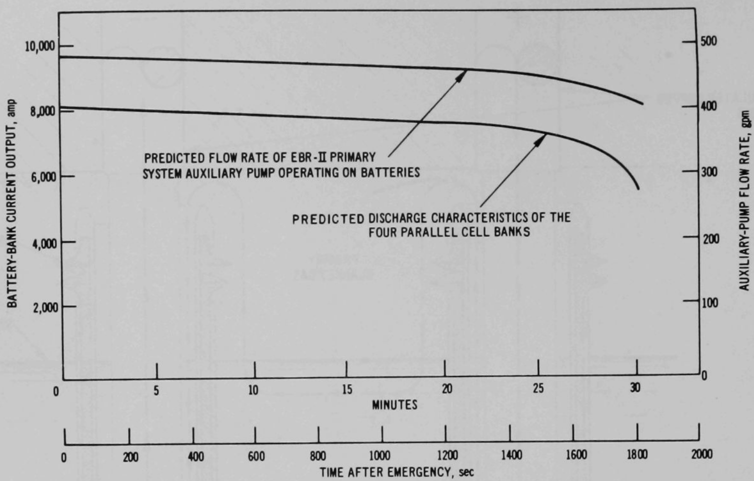


Fig. 6. Flow Characteristics of Auxiliary Pump on Battery Power

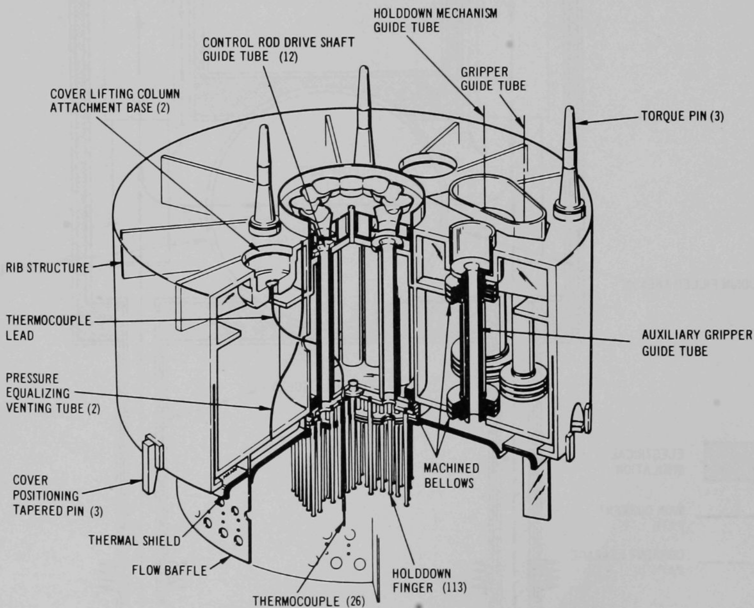
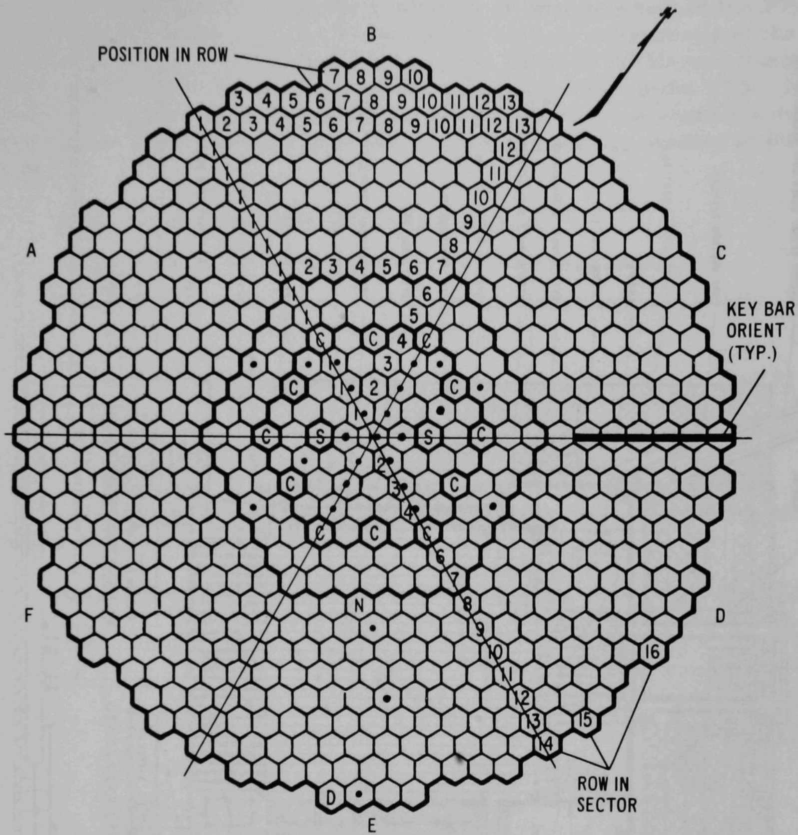


Fig. 7. Reactor-vessel Top Cover



LEGEND

1.	SECTORS	A to F
2.	CONTROL RODS (12)	C
3.	SAFETY RODS (2)	S
4.	THERMOCOUPLES (26)	•
5.	FIXED DUMMY	D
6.	NEUTRON SOURCE	N
7.	GRID POSITIONS	
	CORE	61
	INNER BLANKET	66
	OUTER BLANKET	510
	TOTAL	637

Fig. 8. Subassembly Arrangement in the Reactor

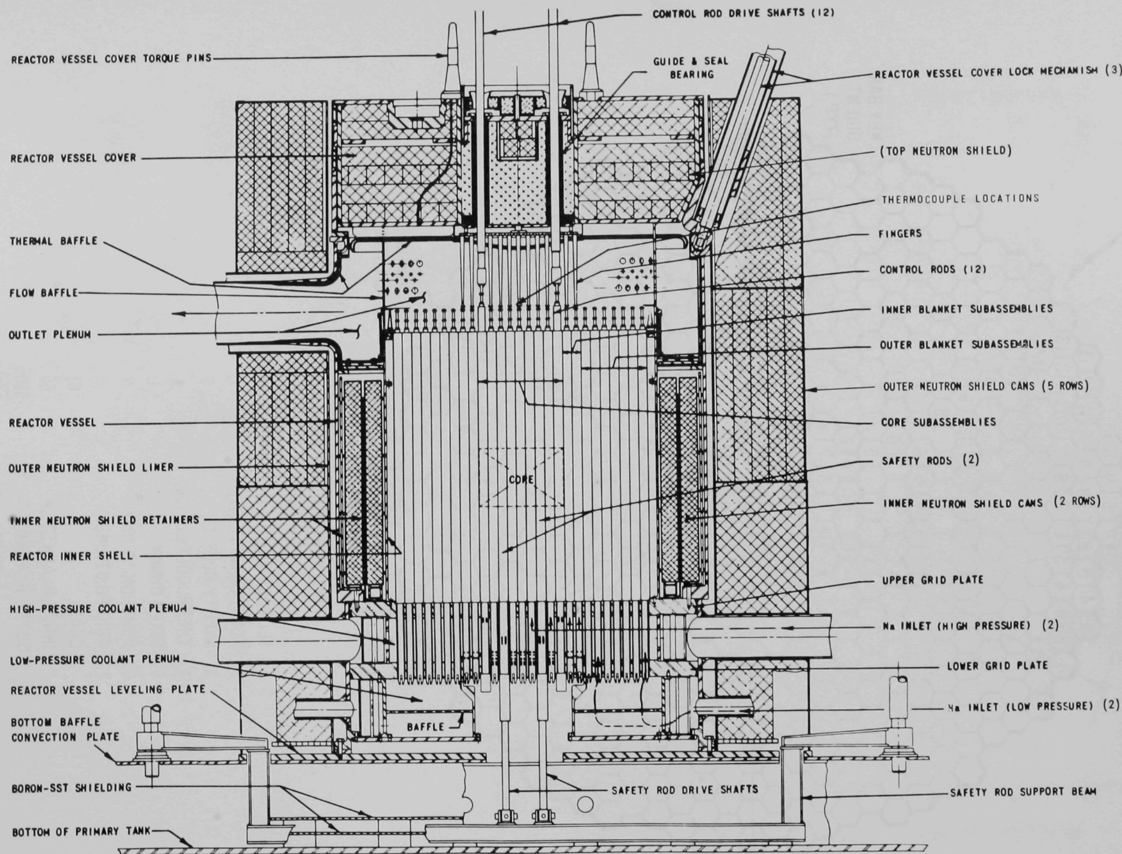


Fig. 9. Reactor-vessel and Neutron-shield Assembly

operating conditions were chosen to provide an adequate test of the PPS circuits and establish a design basis for the various subsections of the PPS. The hypothetical malfunctions of components listed in the Hazard Summary Report (HSR) imply partial or complete loss of portions of the PPS. To provide a broader coverage of the response of the various shutdown modes of the PPS to these malfunctions, the following cases are studied in this report:

Case 1: All shutdown modes operational.

Case 2: A partially inoperative "fuel-handling mode."

Case 3: An inoperative "fuel-handling mode" and a partially inoperative "operating mode."

The "fuel-handling mode" contains nuclear channels 1-3 and 7.

The "operating mode" contains nuclear channels 1-3 and 11, and circuits sensing reactor coolant flow and temperature.

A mode common to both of the above, called the "common mode," contains nuclear channels 4-6.

A detailed discussion of the PPS shutdown modes is given in Section V.

III. TRANSIENT MATERIAL-PERFORMANCE RESTRAINTS

Restraints on transient temperatures of EBR-II core materials were chosen on the basis of physical events that could lead to undesirable conditions in the core. Basically, these events involve phase changes in core materials. These phase changes consist of fuel melting, sodium boiling, or formation of uranium-stainless steel eutectic on the hot spot of cladding on driver-fuel elements. The restraints were thus chosen to be the temperatures at which the following events would occur:

1. Melting of uranium-5 wt % fissium.
2. Formation of uranium-stainless steel eutectic.
3. Cladding rupture.*
4. Melting of experimental oxide fuel.
5. Vaporization of experimental oxide fuel.
6. Melting of experimental carbide fuel.
7. Boiling of sodium coolant.

These temperature restraints, listed in Table I, are believed to be the main considerations in assessing the severity of the consequences following a malfunction of a reactor component. These restraints were used and protective margins were estimated to provide a high degree of assurance that no component malfunction would cause any of the above-mentioned temperatures to be reached (assuming PPS action).

TABLE I. Restraints on Transient Temperatures of Materials

Reactor-core Material	Undesirable Condition	Restraint on Transient Temperature,[a] °F
Driver-fuel cladding	Formation of uranium-stainless steel eutectic	1319
Sodium coolant	Local boiling of sodium coolant	1641
Driver fuel	Melting	1834
Experimental oxide fuel	Melting	5074
Experimental oxide fuel	Vaporization	7200
Experimental carbide fuel	Melting	4800
All cladding material	Rupture (due to pressure or expansion)	1800

[a]Actual temperature at which undesirable condition would occur; no allowance for postulated uncertainties.

*Not considered in this report, but listed to complete the list of restraints.

IV. DYNAMIC SIMULATION OF REACTOR SYSTEMS DURING HYPOTHETICAL MALFUNCTIONS OF COMPONENTS

The dynamic modeling of the response of the EBR-II PPS to component malfunctions requires the simulation of fuel-element heat transfer, reactor kinetics, temperature-induced reactivity feedbacks, component malfunctions, and PPS actions. The AIROS-IIA Dynamic Simulation Digital Code[3] was used in all the following safety studies to combine the thermal, neutronic, and hydraulic characteristics of the reactor system with the postulated malfunctions and the PPS action.

A. Reactor Fuel Elements

Seven fuel-element channels were used to describe the principal EBR-II core and blanket elements. These channels simulated:

1. A peak-temperature driver-fuel element.
2. A feedback driver-fuel element.*
3. A peak-temperature oxide-fuel element.
4. An average oxide-fuel element.
5. An average carbide-fuel element.
6. An inner-blanket depleted-uranium element.
7. An outer-blanket depleted-uranium element.

Figure 10 shows the basic nodal heat-transfer model used to simulate the core and blanket elements. The most important fuel subassemblies in the following studies are the driver-fuel and the experimental oxide-fuel subassemblies. The physical characteristics of these subassemblies are depicted in Figs. 11 and 12. Table II presents the operating parameters of each fuel element used in this study.

B. Temperature-induced Reactivity Feedback Networks

The combined thermal, neutronic, and hydraulic behavior of the reactor system was simulated with the closed-loop feedback diagram depicted in Fig. 13. The reactor-kinetics data used gave excellent agreement with zero-power transfer-function measurements made on EBR-II. Only temperature changes in the feedback driver-fuel and inner-blanket channels were assumed to provide temperature-induced reactivity feedbacks for the entire reactor. All other fuel channels were monitored for consequences during each assumed malfunction of a component.

---

*Designed to simulate the average driver-fuel element in the core and designed to operate at core-averaged temperatures so that reactivity feedback from this channel is typical of entire core.

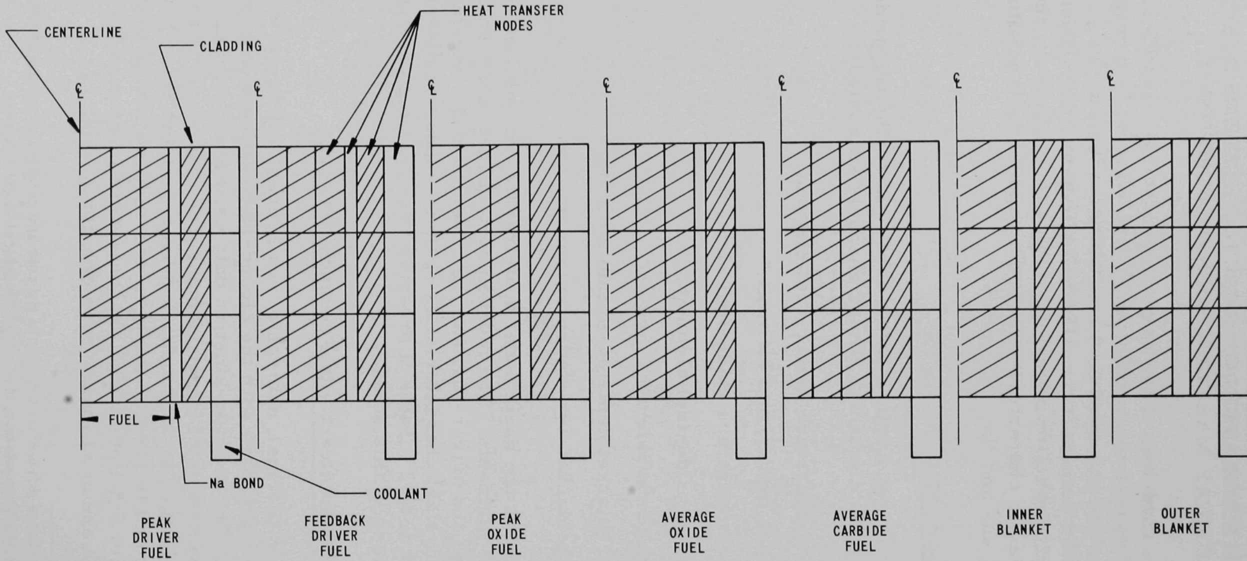


Fig. 10. Seven-channel Dynamic Simulation of EBR-II

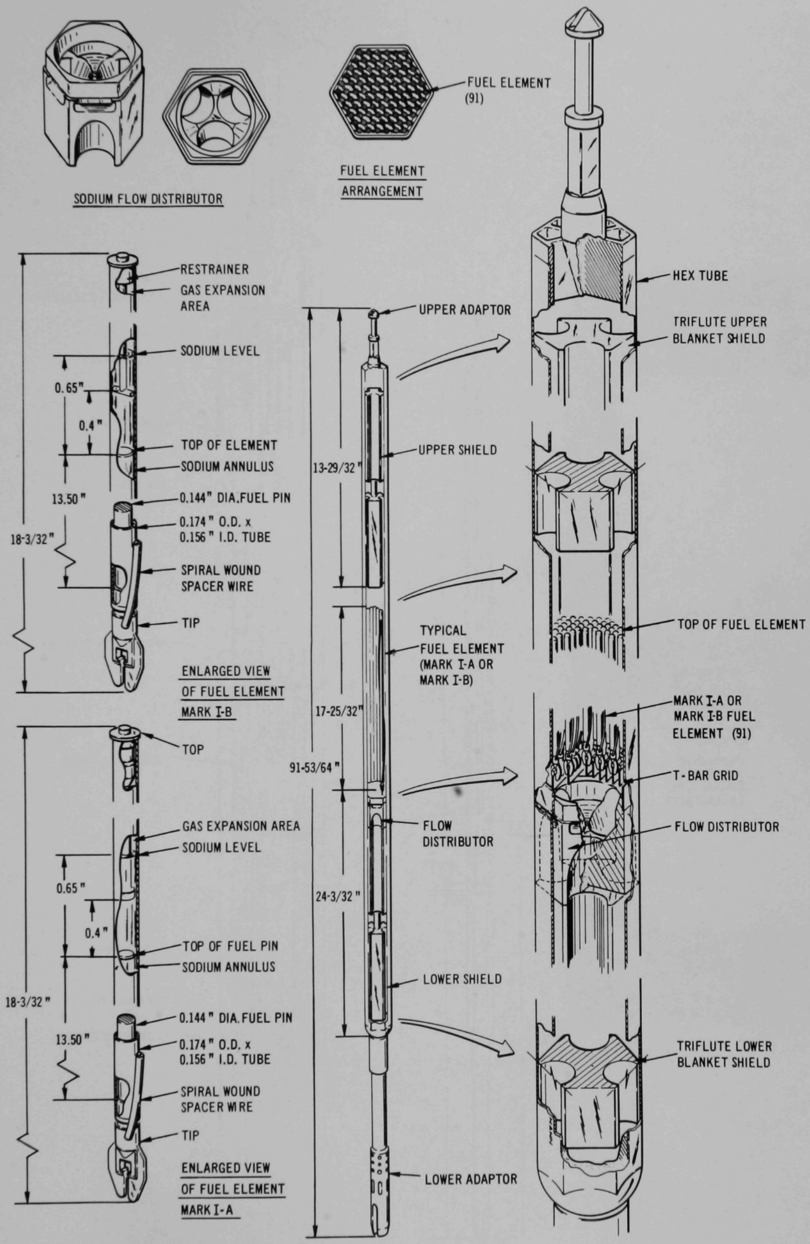


Fig. 11. Mark-1A and -1B Model STB Fuel Subassemblies

The core- and blanket-averaged temperature coefficients used in this study to simulate driver-fuel and blanket axial expansion and sodium-density changes are similar to the prompt negative feedback parameters used successfully in predicting the transient response of EBR-II cores to various rod-drop experiments.[4]

C. Component Malfunctions

The technical approach in this report is to overlay the setpoints and response times of the EBR-II PPS onto the seven hypothetical malfunctions of reactor-system components listed in the Hazard Summary Report. The various simultaneous occurrences required so that the component malfunctions cited in the HSR lead to potentially hazardous situations are not credible; they are only used in this report to formalize the design basis and the operating range for those portions of the PPS circuits sensing reactor parameters of flux, flow, and temperature.

Three types of components are involved:

1. Control and safety rods.
2. Fuel-handling mechanisms.
3. Primary coolant pumps.

Abnormal operations of the above components could adversely affect the flux, flow, and temperatures of the reactor system. The response of the EBR-II PPS can therefore be tested by assuming various combinations of startup and at-power conditions involving the above-mentioned hypothetical malfunctions.

The dynamic modeling of each component malfunction leading to a reactivity change was simulated by assuming a linear reactivity insertion. The reactivity ramp rates and total worths of components are presented in Table III. The response of the PPS to any of the above malfunctions was simulated in the AIROS-IIA code by establishing a trip setpoint and an associated response time with respect to operating parameters in the feedback driver-fuel channel. The subsequent action and movement of the safety or control rods were simulated by assigning a delay for the decay of the magnetic field holding the rods and by assigning an appropriate weighing function to simulate the axial reactivity-worth curve for the rods as they were tripped out of the reactor core. All circuit-response times, safety- and control-rod delays, and rod-acceleration curves during a reactor trip were taken from in situ measurements made at the EBR-II reactor plant. PPS response-time measurements are presented in Appendix B.

TABLE III. Hypothetical Malfunctions Used to Study the Response of the EBR-II PPS

No.	Type of Malfunction	Relative Neutron Level	Reactivity Worth of Trip Rods,[a] $	Reactivity Ramp, $/sec	Total Reactivity Worth of Component, $
1.	Safety rods driven into reactor core	10^{-4}	-1.36	0.0067	1.36
2.	Central subassembly driven into reactor core at low speed 6	10^{-4}	-1.36	0.0134	1.65
3.	Single control rod driven into reactor core during startup	10^{-4}	-5.00	0.0056	0.57
4.	Single control rod driven into reactor during full-power operations	1.0 (62.5 MWt)	-5.00	0.0056	0.57
5.	Dropping of central subassembly into reactor core during fuel handling	10^{-4}	-1.36	3.7500	1.65
6.	Central subassembly driven into reactor core at high speed 72	10^{-4}	-1.36	0.1600	1.65
7.	Loss of primary pumping during full-reactor-power operations	1.0 (62.5 MWt)	-5.00	-	-

[a]The lower value represents a situation in which only the safety rods are available for trip; the higher value represents a situation in which both control and safety rods are available.

D. Initial Conditions for Dynamic Analysis

Many of the hypothetical malfunctions listed in the HSR inherently contain the assumption of a partially inoperative PPS. Moreover, it must be assumed that portions of the PPS are inactivated so that a design basis for circuit performance can be established. Under conditions of a fully operative PPS, including all auxiliary circuits not sensing reactor parameters, all malfunctions of components would be sensed before any appreciable change in reactor operating parameters. To avoid such a limitation, the following dynamic studies use three different combinations of operating conditions to establish the design basis for the operating range of the flux, flow, and temperature portions of the PPS. The combinations of nuclear, flow, and temperature channels assumed to be operational in each phase of the study are shown in Table IV. Case 1 is listed merely to indicate that there are additional auxiliary circuits not sensing reactor operating parameters that will provide early protection against component malfunctions. This report is devoted to establishing the design basis for those portions of the PPS sensing the reactor operating parameters of flux, flow, and temperatures, and therefore covers only cases 2, 3, and 4.

TABLE IV. Combinations of PPS Action in Response to Various Hypothetical Malfunctions of Components

Case No.	Case Conditions	Type of Malfunction	Condition of PPS	Nuclear Channels[a] Period Trip 1	2	3	4	5	6	Level Trip 1	2	3	7	9	10	11	Outlet-coolant-temperature Trip	Change-in-flow Trip
1	Normal operations	"Normal performance"	All shutdown modes operational (including auxiliary circuits)	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
2	Normal operations	Reactor and/or primary system	All shutdown modes operational (not including auxiliary circuits)	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
3	Fuel handling	Reactor and/or fuel handling	Partially inoperative fuel-handling mode	-	-	-	x	x	x	-	-	-	-	-	-	-	-	-
4	Startup	Reactor controls	Inoperative fuel-handling mode and no period protection in operate mode	-	-	-	-	-	-	-	-	-	-	x	x	x	x	x

[a] x Available.
- Not available.

Case 2 is designed to allow an assessment of all shutdown protective modes during component malfunctions, assuming that all auxiliary circuits are inoperative. Results from this case will aid in establishing the "normal" PPS response to hypothetical malfunctions of components.

Case 3 is designed to test the response of the PPS assuming that all auxiliary circuits and some portions of the fuel-handling modes are inoperative. Results from this case will demonstrate the action of redundant backup in the PPS.

Finally, case 4 is designed to test the response of the PPS, assuming that all auxiliary circuits, all of the fuel-handling mode, and all of the reactor period-trip protection are inoperative. Case 4 is the most severe test of the response of the PPS and is presented as an extreme upper boundary for PPS performance.

All PPS response terminates in the activation of the control and/or safety rods. The control rods are activated only in the "operate mode" by the PPS. During fuel handling, the control rods are disconnected so that only the safety rods can be activated. The control-rod locations are shown in Fig. 1. Figure 14 gives a typical displacement curve for the control rods following a trip signal. Figure 15 presents a typical displacement curve for the safety rods. Figure 16 shows the locations and means of movement of the EBR-II safety rods. The effectiveness of the PPS is studied in the following sections.

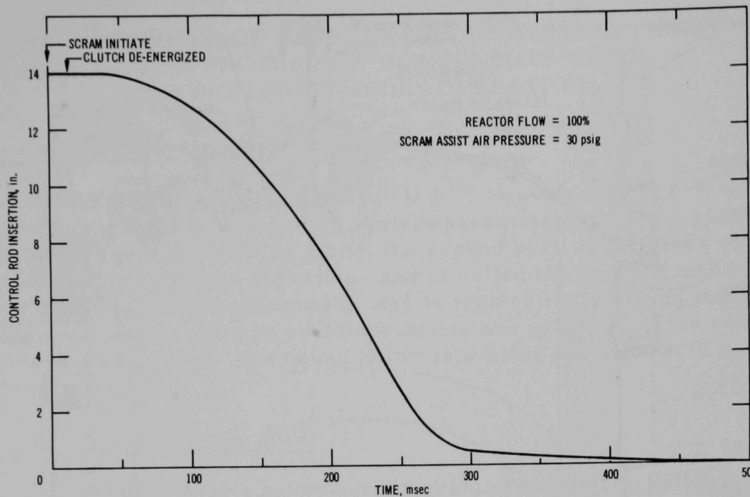


Fig. 14. Typical Time/Displacement Curve for a Control Rod during Scram

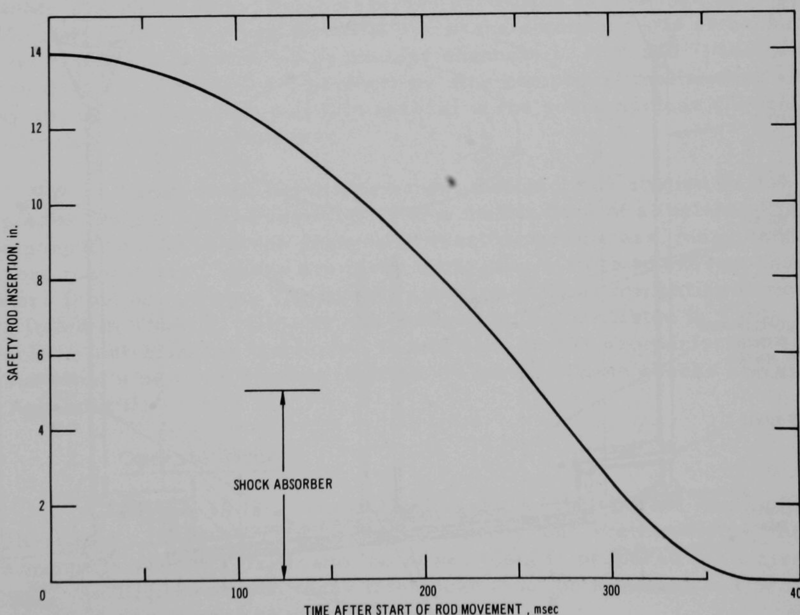


Fig. 15. Safety-rod Displacement vs Time after Start of Rod Movement

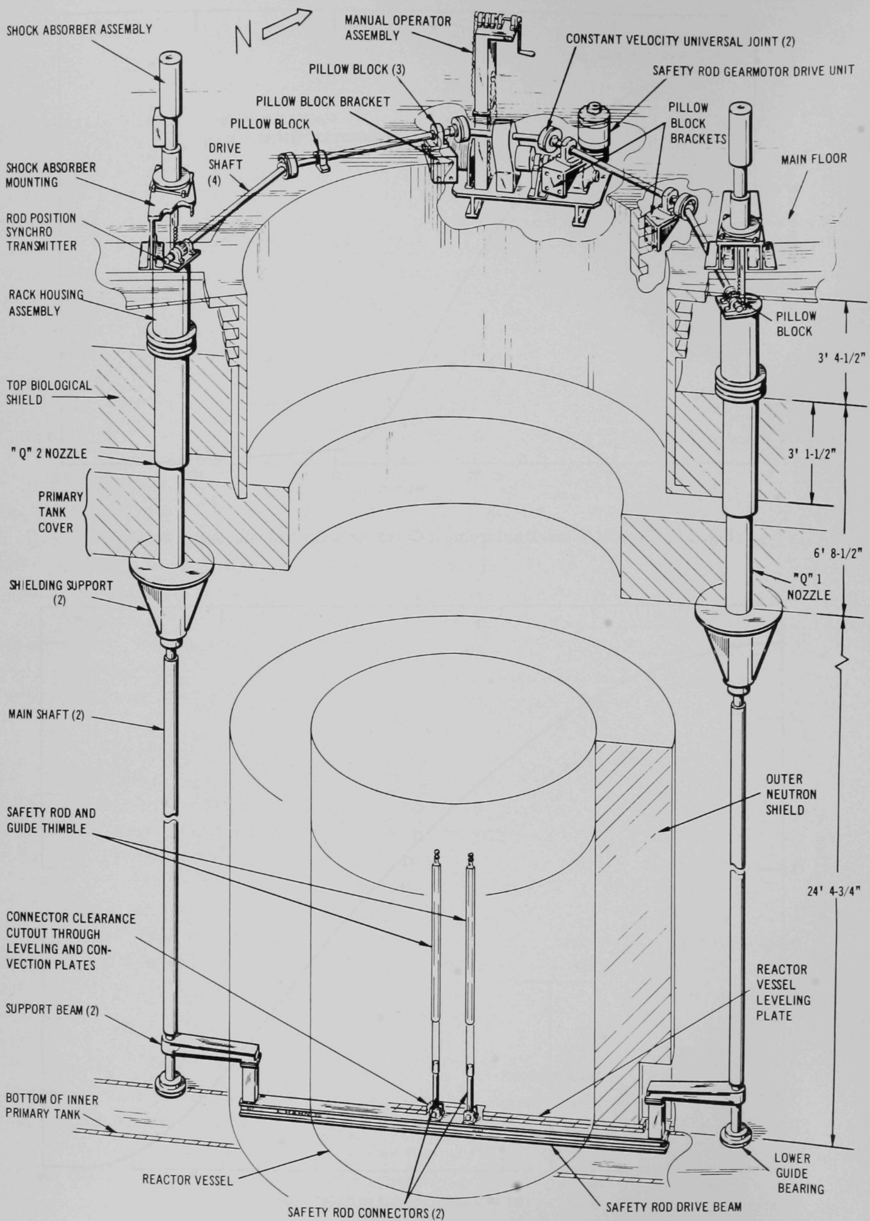


Fig. 16. Safety-rod Drive Assembly

V. RESPONSE OF CURRENT PPS TO HYPOTHETICAL MALFUNCTIONS OF COMPONENTS WITH ALL SHUTDOWN MODES OPERATIONAL

A. Reactor-shutdown Schematic Diagrams

As mentioned earlier, the EBR-II PPS is composed of two shutdown modes--the first portion monitors system parameters during fuel handling and is called the fuel-handling mode; the second portion monitors system parameters during power operations and is called the operate mode. The portion common to both shutdown modes is termed the common mode. The response and action of these shutdown modes are analyzed in the remaining sections of this report. Presented below is a brief description of each shutdown mode.

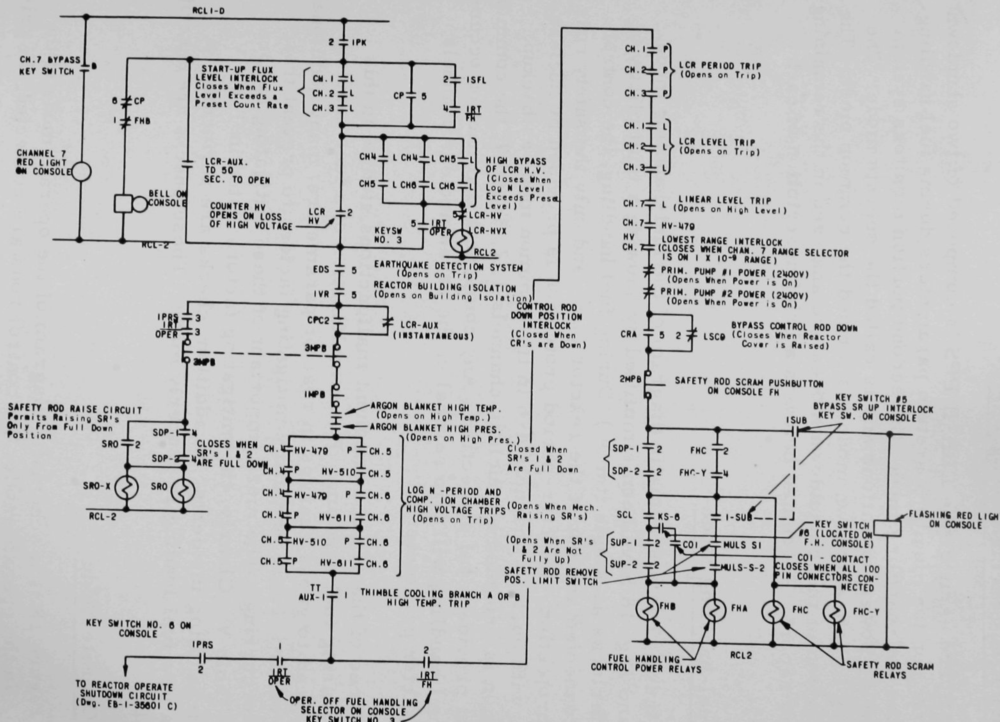
1. Fuel-handling Mode

Figure 17 is a schematic diagram of the fuel-handling mode. (This figure and Fig. 18 have been simplified by showing a multiplicity of relays and interlocks as single items.) During fuel handling, the control rods are disconnected and out of the reactor core, and only the safety rods provide trip protection. Reactor-period protection is provided by nuclear channels 1-3, with nuclear channels 4-6 in the common mode as backup. Level protection is provided by nuclear channels 1, 2, 3, and 7; the common mode does not provide level protection. Any component malfunction occurring during fuel handling will trip several of the seven nuclear channels monitoring reactor parameters.

Three of the seven component malfunctions studied in this report occur during fuel handling. For a malfunction of a fuel-handling component to lead to gross changes in reactor parameters, many additional errors must occur. There are many mitigating factors to prevent these errors from occurring. The more important of these mitigating factors are listed in Table V. Clearly, the mitigating factors listed in Table V preclude fuel-handling incidents. In addition, checks are performed to ensure that a new fuel loading is properly made. These checks are described in Appendix D.

2. Operate Mode

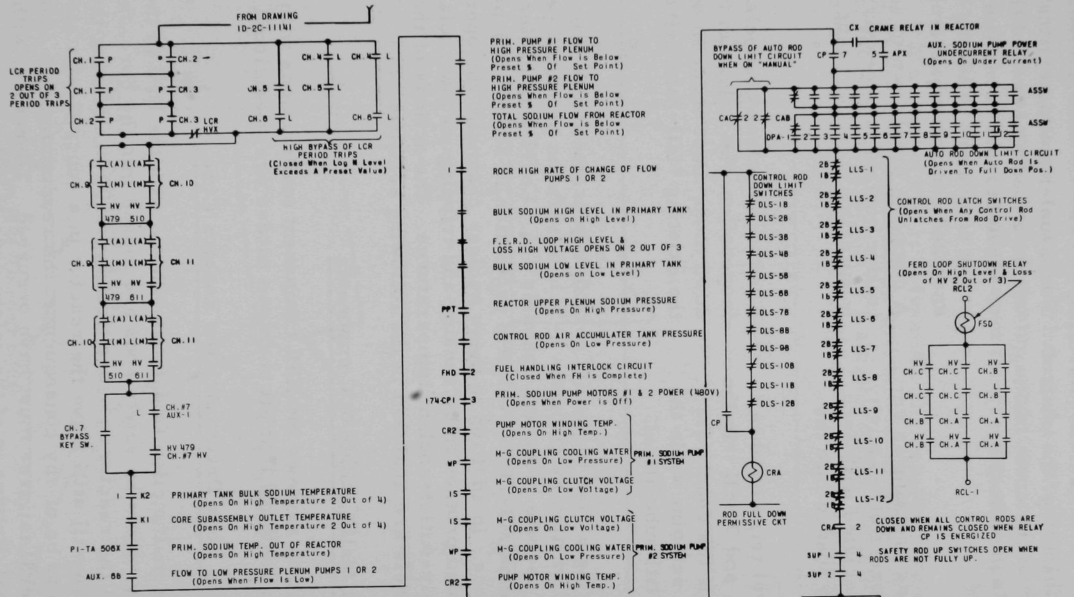
Figure 18 is a schematic diagram of the operate mode of the EBR-II PPS. In the operate mode, the control rods are connected, the primary pumps are energized, and the power plant is prepared for a rise to power. As Fig. 18 shows, many interlocks must be satisfied in a prescribed sequence before a rise to power is permitted.



NOMENCLATURE

1MPB - MANUAL SHUTDOWN PUSHBUTTON (CONSOLE)
2MPB - MANUAL SHUTDOWN PUSHBUTTON (FH CONSOLE)
3MPB - SAFETY-ROD SCRAM PUSHBUTTON (CONSOLE)
CP - CONTROL-POWER "ON" RELAY
CR - CONTROL ROD
CRA - INTERLOCK RELAY (CONTROL ROD DOWN)
EDS - EARTHQUAKE DETECTION SYSTEM
FH - FUEL HANDLING
FHA - FUEL-HANDLING CONTROL-POWER RELAYS
FHB - FUEL-HANDLING CONTROL-POWER RELAYS
FHC - SAFETY-ROD RAISE RELAYS (FH MODE)
FHC-Y - SAFETY-ROD RAISE RELAYS (FH MODE)
HV - HIGH-VOLTAGE RELAY
HVX - HIGH-VOLTAGE AUXILIARY RELAY
IPRS - REACTOR STARTUP PERMISSIVE SWITCH ON CONSOLE
IRT/FH - KEYSWITCH FOR FUEL-HANDLING MODE
IRT/OPER - KEYSWITCH FOR OPERATE MODE
IVR - REACTOR-BUILDING ISOLATION
LCR - LOG COUNT RATE
MULS - MAXIMUM UP LIMIT (SAFETY ROD)
RCL - POWER BUS
SDP - POSITION RELAY (SAFETY ROD DOWN)
SR - SAFETY ROD
SRO - SAFETY-ROD RAISE RELAYS
SRO-X - SAFETY-ROD RAISE RELAYS
SUP - POSITION RELAY (SAFETY ROD UP)
TT - THIMBLE-COOLING TRIP

Fig. 17. Schematic Diagram of Fuel-handling Mode



NOMENCLATURE

APX	-	POWER PERMISSIVE (AUX. PRIMARY SODIUM PUMP)
ASSW	-	CONTROL-ROD AUTO SEL. SWITCH
CAB	-	CONTROL-ROD AUTO OPER. RELAY
CAC	-	CONTROL-ROD AUTO OPER. RELAY
CP	-	CONTROL POWER "ON" RELAY
CPA	-	CONTROL POWER "ON" RELAY
CPC	-	CONTROL POWER "ON" RELAY
CX	-	CRANE OVERTRAVEL RELAY
CY	-	CRANE OVERTRAVEL RELAY
DPA	-	POSITION RELAY (CONTROL ROD DOWN)
FBP1	-	FLUX-TRIP BYPASS RELAY
FBP2	-	FLOW-TRIP BYPASS RELAY
F.E.R.D.	-	FUEL-ELEMENT RUPTURE DETECTOR
HV	-	HIGH-VOLTAGE RELAY
HVX	-	HIGH-VOLTAGE AUXILIARY RELAY
L(A)	-	AUTOMATIC FLUX-LEVEL RELAY
L(M)	-	MANUAL FLUX-LEVEL RELAY
LCR	-	LOG COUNT RATE
LLS	-	CONTROL-ROD LATCH SWITCH
M-G	-	MOTOR-GENERATOR SET
RCL	-	POWER BUS
RHX	-	PRIMARY-PUMP DEVIATION RELAY
ROCR	-	RATE-OF-CHANGE RELAY
SUP	-	POSITION RELAY (SAFETY ROD UP)

Fig. 18. Schematic Diagram of Operate Mode

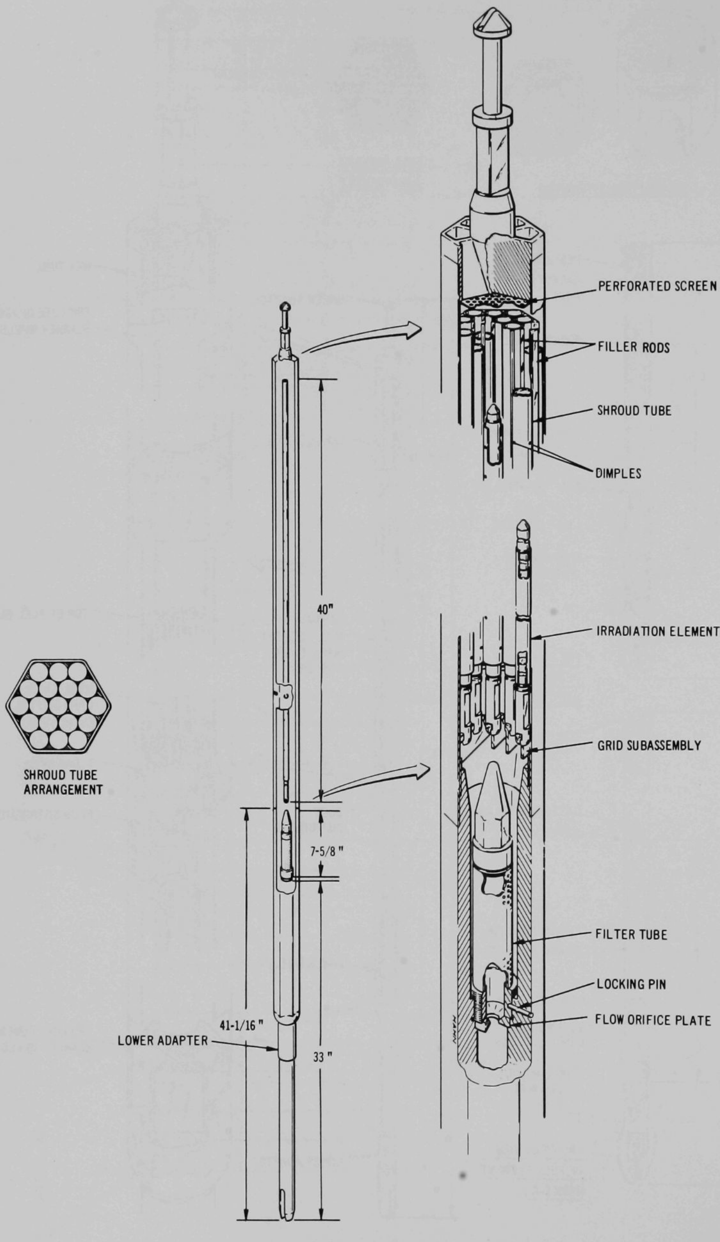


Fig. 12. Mark-A Irradiation Subassembly

TABLE II. Physical Characteristics of Typical EBR-II Fuel and Blanket Channels

Type of Channel	Active Height, ft	Diameter of Fuel or Blanket Pin, in.	Power per Channel, Btu/sec	Mass Flow Rate per Channel, lb/sec
Peak driver fuel	1.125	0.144	9.832	0.1490
Feedback driver fuel	1.125	0.144	7.548	0.0980
Peak experimental oxide fuel	0.917	0.250	18.00	0.210
Average experimental oxide fuel	1.125	0.250	13.60	0.1550
Average experimental carbide fuel	1.167	0.300	22.70	0.3140
Inner-blanket element	4.600	0.433	12.70	0.1280
Outer-blanket element	4.600	0.433	1.945	0.0362

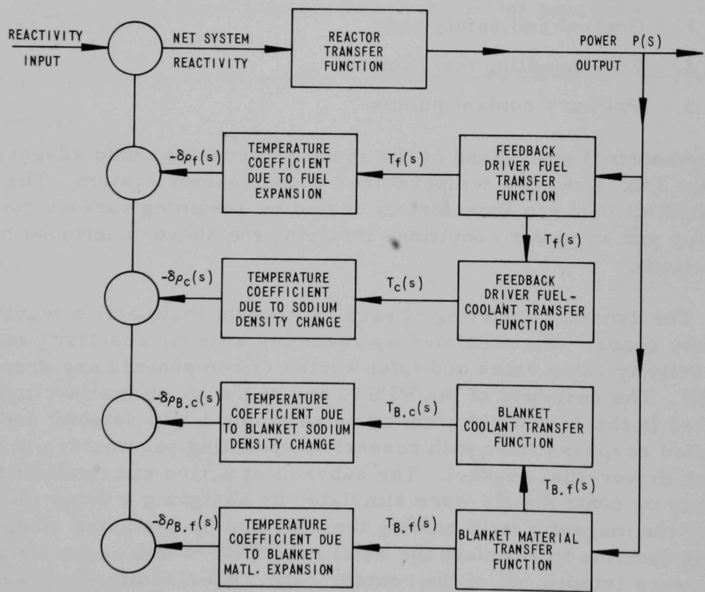


Fig. 13. Power-to-Reactivity Feedback Network

TABLE V. Mitigating Factors with Respect to Fuel-handling Incidents

Mitigating Factor	Comment
1. During fuel handling, the control rods are out of the core, making the reactor subcritical by -1500 Ih.	For a fuel-handling criticality to occur, a loading error of the order of +1500 Ih must occur (i.e., an error equivalent to adding five fully enriched central subassemblies).
2. Fuel-handling procedures require monitoring of reactivity changes associated with each subassembly loaded into the core.	A complete breakdown of administrative control is necessary for this monitoring to fail to occur.
3. Nuclear-level trips on channels 1, 2, 3, and 7 will trip the safety rods if the count rate exceeds the setpoint while the reactor is still subcritical.	For these nuclear channels to be inoperative, the failure in the shutdown mode must occur in spite of the existing "fail-safe" circuitry.

During a rise to power, period-trip protection is provided for a short interval of time by channels 1-3; however, the principal period channels in the operate mode are channels 4-6. Level-trip protection is provided by channels 9-11. Level-trip protection is set for full-power reactor conditions. In addition to these nuclear channels, the operate mode provides for added protection by sensing and tripping on low coolant flow or high subassembly-outlet-coolant temperatures. The actions of the above-mentioned channels are analyzed in the following sections.

B. Hypothetical Malfunctions of Components

This section reviews the HSR component malfunctions with respect to case 2, which assumes that all shutdown modes are operational.

1. Safety Rods Driven into the Reactor Core

This malfunction is only considered with the PPS in the operate mode. Normally, when the safety rods are moved into the core, the control rods are out and the reactor is subcritical by -1500 Ih. Moving the safety rods into the core will still leave the reactor in a subcritical state. For this malfunction to be of any consequence, the reactor must be inadvertently critical and several failures must have occurred. Therefore, with all PPS modes operational, driving in of the safety rods is of little consequence.

2. Control Rod Driven into the Reactor Core

This malfunction can only occur with the PPS in the operate mode, since the control rods are physically disconnected in the fuel-handling mode. In the operate mode with all interlocks satisfied, full coolant flow is provided to the reactor. Under startup conditions, reactor-period trips on channel 1, 2, 3, 4, 5, or 6 will protect the reactor with no change in startup isothermal conditions.

If a control rod is driven into the reactor core at full power, the operating power and temperatures will increase. Figure 19 shows the results of inadvertent insertion of a single control rod into a core operating at full power and full coolant flow with a power-level trip setpoint at 110% of

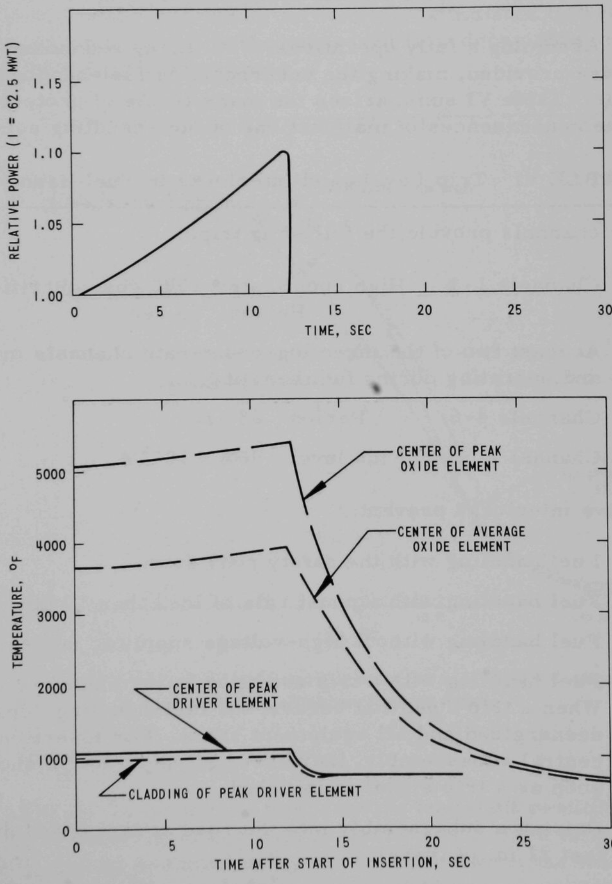


Fig. 19. Power and Temperature Curves following Driving of Single EBR-II Control Rod into Core at 62.5 MWt; Power-level Trip Setpoint at 110% of Full Power

full power. Notice the sharp break following the reactor trip, indicating that the prompt response of the PPS and the speed and action of the control rods is more than sufficient to terminate the power increase at the prescribed power-level setpoint. During this overpower condition, the peak oxide, the average oxide, and the peak driver-fuel center temperatures are the most important reactor operating parameters. The peak driver-fuel cladding and peak driver-fuel center temperatures shown in Fig. 19 are well within the performance capabilities of these reactor materials. Prompt, responsive protection is available for an at-power control-rod malfunction, assuming a control-rod bank worth of about 5.00$ and a response time of about 50 msec (i.e., the total delay between the receiving of the trip signal and the moving of the control rods).

3. Malfunctions of Fuel-handling Components

Assuming a fully operational PPS, many redundant levels of protection are provided, making the occurrence of fuel-handling incidents not plausible. Table VI summarizes the many levels of protection available to negate the consequences of malfunctions of fuel-handling components.

TABLE VI. Trip Levels and Interlocks in Fuel Handling

1. Nuclear channels provide the following trips:

   a. Channels 1-3 — High count rate: 1500 cps subcritical; Period: 25 sec

      At least two of the three log-count-rate channels must be on and operating during fuel handling.

   b. Channels 4-6 — Period: 25 sec

   c. Channel 7 — Flux level: 96% x 10^{-9} A

2. Protective interlocks prevent:

   a. Fuel handling with the safety rods down.

   b. Fuel handling with a count rate of less than 10 cps.

   c. Fuel handling without high-voltage supply to nuclear detectors.

   d. Fuel handling with a trip condition in the shutdown circuits. When a trip condition occurs, the fuel-handling console is deenergized and all equipment stops. For insertion of the central subassembly, therefore, gripper motion should stop as soon as a trip signal is initiated.

   e. Driving a subassembly into the core at fast speed during the last 21 in. of travel.

   f. Opening the gripper jaws at an improper time.

C. Loss of Primary Pumping Power

A loss of primary pumping power at full reactor power of 62.5 MWt will cause a decrease in primary coolant flow and lead to an increase in coolant-outlet temperature. Both of these parameters have trip protection and will activate the EBR-II control rods. This malfunction has been studied, and the results are presented below.

Figure 20 shows the measured and simulated primary-coolant flow decay following the loss of primary pumping power. The dynamic simulation program AIROS-IIA allows only a parametric equation to be used to simulate flow decay. To ensure that the measured flow decay was bracketed, two curve fits were made, as shown in Fig. 20. One, termed the "simulated HSR flow decay," is an approximation to the flow decay presented in the Addendum to the HSR. The other, termed the "simulated measured flow decay," is an approximation to the measured flow decay obtained in early tests on the EBR-II primary pumps.

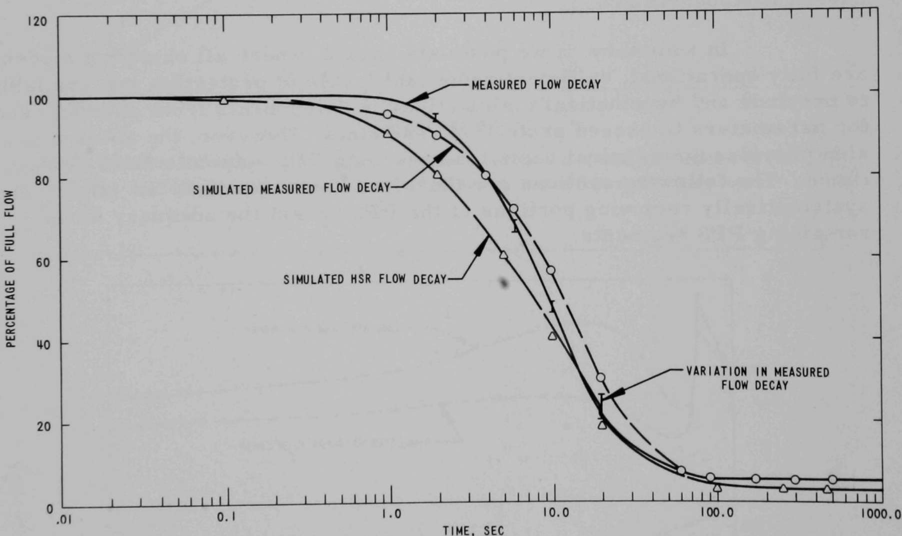


Fig. 20. Measured and Simulated Curves for Decay of Primary Coolant Flow following Loss of Primary Pumping Power

1. Temperature Trip

Figure 21 shows the temperature transient resulting from a loss of primary pumping power with protection by a fixed temperature setpoint of 940°F and an assumed delay of 10 sec (between the loss of pumping power and the trip). The peak driver-fuel cladding temperatures probably lie between the two curves presented. Several time delays were studied, and the results of these studies are presented in Fig. 22. Using the "simulated HSR flow decay," the uranium-stainless steel eutectic temperature is reached with delays of 15 sec or more; using the "simulated measured flow decay," the eutectic temperature is reached with delays of 23 sec or more. (Here it is assumed that a temperature trip is the only protection against this accident.)

2. Low-flow Trip

On the assumption that a loss of primary pumping power is responded to by a low-flow trip at 94% of full flow, Fig. 23 indicates the resulting peak driver-fuel cladding temperatures for various delays in response. The peak temperatures are lower with sensing of reduced flow than with sensing of high coolant-outlet temperature. These results are summarized in Section VIII, along with protective margins and their use in establishing operating ranges for the PPS. (Response of the PPS to other abnormal conditions, such as sudden stoppage of a primary pump, is discussed in Appendix E.)

In summary, if we postulate case 2, where all shutdown modes are fully operational, sufficient redundant levels of protection are available to preclude any hypothetical malfunctions of components from causing reactor parameters to exceed protective restraints. However, the above discussion provides no technical basis for assessing PPS setpoints and response times. The following sections are therefore concerned with the effects of systematically removing portions of the PPS to test the adequacy of the remaining PPS segments.

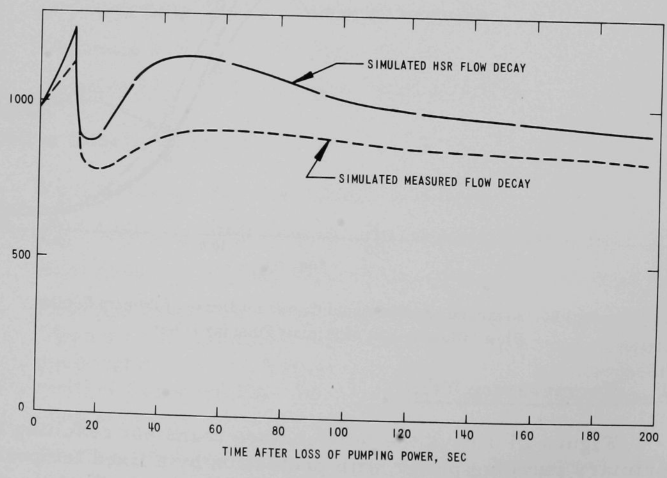


Fig. 21. Peak Driver-fuel Cladding Temperatures following Loss of Primary Pumping Power; Only Coolant-outlet-temperature Trip Protection with 10-sec Delay

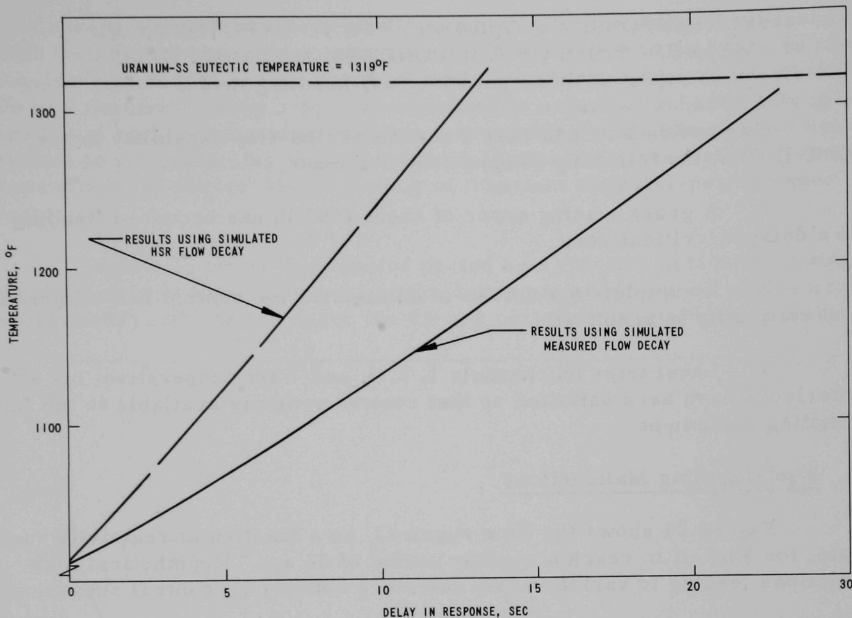


Fig. 22. Peak Driver-fuel Cladding Temperatures as a Function of Delay in Response of PPS to High Coolant-outlet Temperature, following Loss of Primary Pumping Power

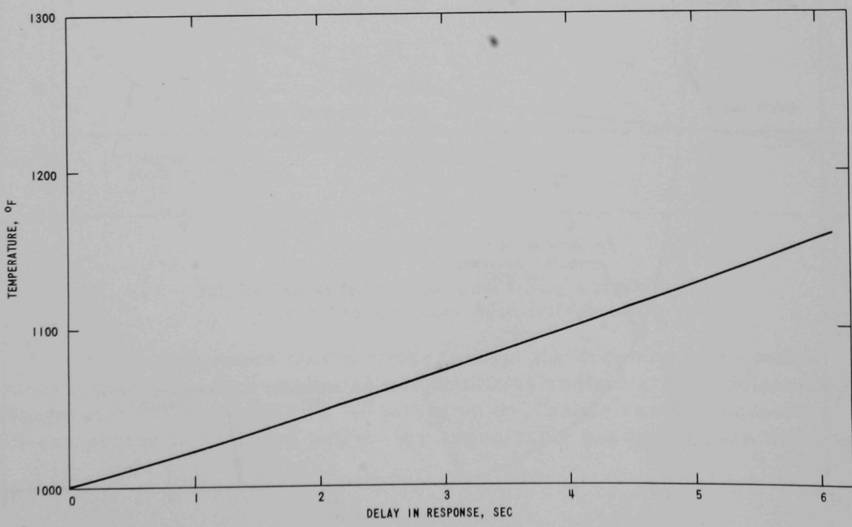


Fig. 23. Peak Driver-fuel Cladding Temperatures following Loss of Primary Pumping Power, Only 6%-low-flow Trip Protection

VI. RESPONSE OF CURRENT PPS TO HYPOTHETICAL MALFUNCTIONS OF COMPONENTS, ASSUMING CASE 3 (Partially inoperative fuel-handling mode)

To provide a test in case 3 for the period-trip capability in the EBR-II PPS, the following assumptions are made:

1. A gross loading error of about 1500 Ih has occurred, leading to a delayed-critical core.

2. A complete breakdown in administrative control has occurred, and criticality is unnoticed.

3. Level trips on channels 1, 2, 3, and 7 are inoperative, but all interlocks have been satisifed so that control power is available to the fuel-handling equipment.

A. Fuel-handling Malfunctions

Figure 24 shows the time required, as a function of reactivity ramp rate, for EBR-II to reach a reactor period of 25 sec. Hypothetical malfunctions leading to various ramp rates are noted; i.e., central subassembly

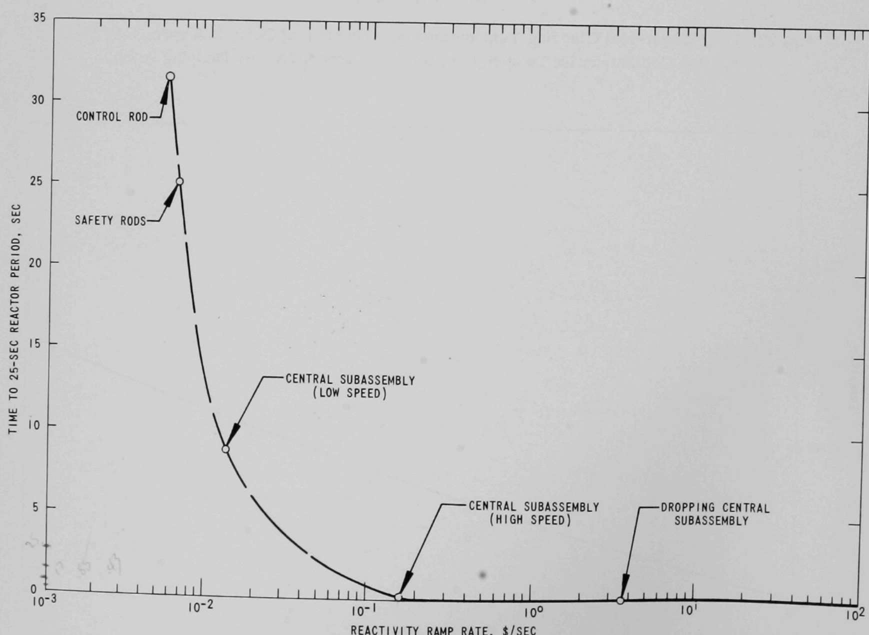


Fig. 24. Time Required to Reach a 25-sec Period-trip Setpoint with Various Ramp Insertions of Reactivity

(low speed), central subassembly (high speed), and dropping of subassembly. Also included are ramp rates for control- and safety-rod insertion to show required times to a 25-sec period if these events are assumed possible. Up to a reactivity ramp rate corresponding to a central-subassembly insertion at high speed, sufficient time is available for the plant protective system to act (assuming the response times measured and listed in Appendix B) to prevent the exceeding of transient material-performance restraints.

Figure 25 shows the reactor period as a function of time following the driving in of the safety rods or the driving in at low speed of a central subassembly. (Times to reach the 25-sec period-trip setpoint are noted.)

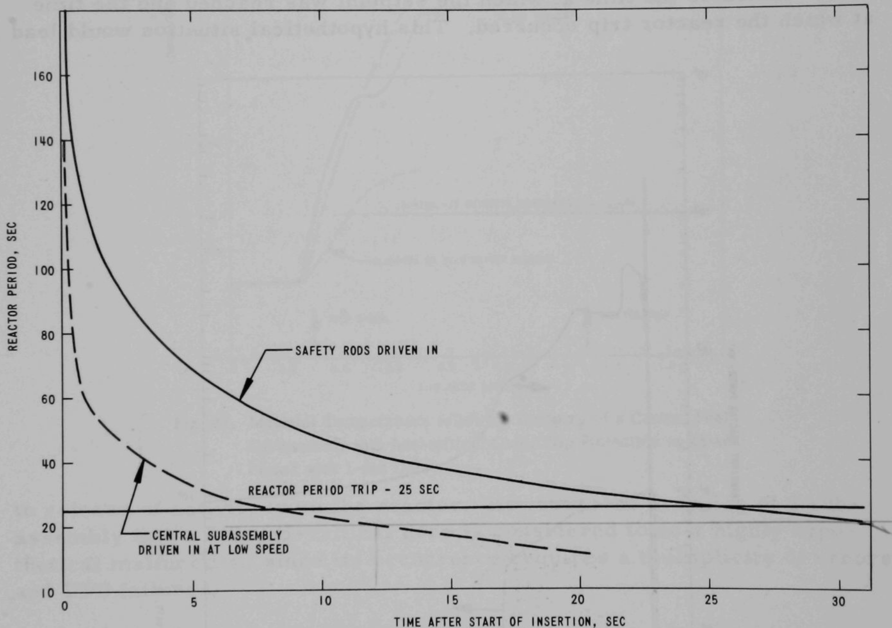


Fig. 25. Reactor Period following the Driving in of EBR-II Safety Rods or a Central Driver-fuel Subassembly

In all component malfunctions, except the dropping of a central subassembly, the core-material temperatures remain at the coolant-inlet temperature of about 700°F. No temperature change results, because a 25-sec period is achieved before any appreciable heating of core materials.

B. Malfunction of Fuel-handling Mechanism

In the extreme case of the dropping of a central driver-fuel subassembly, the measured response times of 0.3-1.5 sec are too long to prevent abnormally high material temperatures from occurring in the fuel elements. Figure 26 shows the resulting power and reactivity distribution following an inadvertent dropping of the central subassembly into a just-critical core. Note that the reactor is prompt critical for a short time. The prompt temperature-induced reactivity feedbacks from fuel and coolant quickly reduce the reactivity of the system, and a reactor trip finally occurs, terminating the transient. Figure 27 shows the transient temperatures following the dropping of a central driver-fuel subassembly. Figure 27 indicates the time at which the setpoint was reached and the time at which the reactor trip occurred. This hypothetical situation would lead

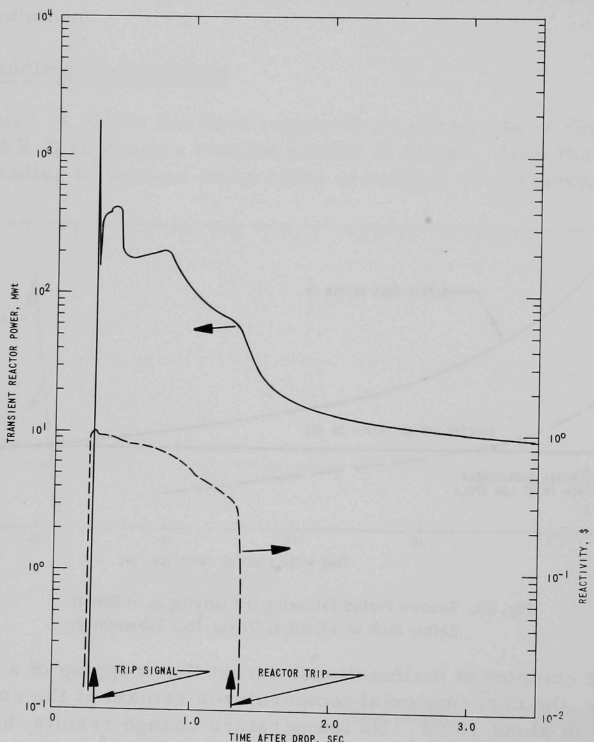


Fig. 26. Power and Reactivity Curves following Dropping of a Central Driver-fuel Subassembly into Just-critical Core; Trip Protection at 25-sec Period with 1-sec Delay

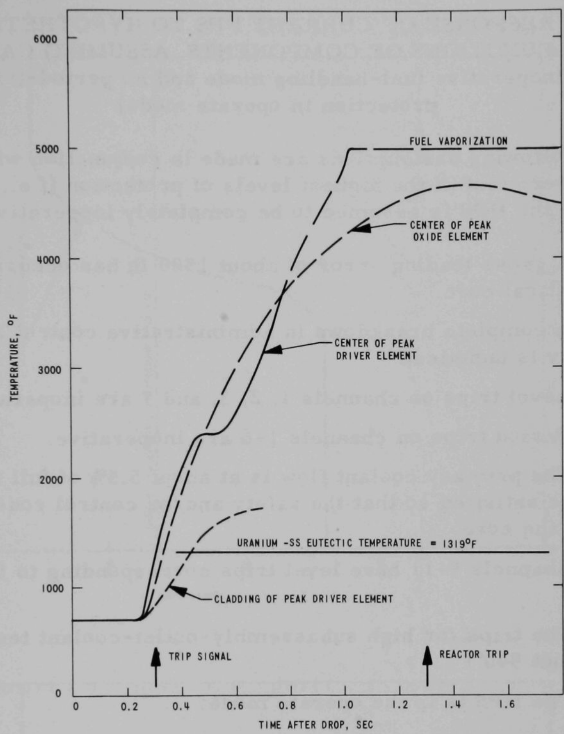


Fig. 27. Material Temperatures following Dropping of a Central Fuel Subassembly into Just-critical Core; Trip Protection at 25-sec Period with 1-sec Delay

to release of activity from the reactor. However, the dropping of a subassembly into a delayed-critical core is considered to be a highly hypothetical malfunction, since its occurrence requires a multiplicity of errors and PPS failures.

In summary, the present PPS provides period-trip protection for all component malfunctions, except the dropping of a central subassembly. (To provide protection against the consequences of dropping a driver-fuel subassembly, response times in the period circuits of less than 50-100 msec are required.)

VII. RESPONSE OF CURRENT PPS TO HYPOTHETICAL MALFUNCTIONS OF COMPONENTS, ASSUMING CASE 4
(Inoperative fuel-handling mode and no period-trip protection in operate mode)

The following assumptions are made in conjunction with case 4 to provide a severe test of the highest levels of protection (i.e., the last levels before the PPS is assumed to be completely inoperative).

1. A gross loading error of about 1500 Ih has occurred, leading to a delayed-critical core.

2. A complete breakdown in administrative control has occurred, and criticality is unnoticed.

3. Level trips on channels 1, 2, 3, and 7 are inoperative.

4. Period trips on channels 1-6 are inoperative.

5. The primary coolant flow is at about 5.5% of full flow, but all interlocks are satisfied so that the safety and/or control rods can be inserted into the core.

6. Channels 9-11 have level trips corresponding to 62.5-MWt operation.

7. The trips for high subassembly-outlet-coolant temperature are set at about 940°F.

8. The PPS is in the operate mode.

A. Safety Rods

1. Power-level Trip

Figure 28 shows the resulting power and reactivity increase following the inadvertent driving of the EBR-II safety rods into a just-critical core under conditions of reduced flow. Note that no reactor trip occurs, since the temperature-induced reactivity feedback in the system due to high fuel and coolant temperatures is very strong and prevents the reactor from reaching the setpoint in nuclear channels 9-11. The maximum system reactivity reaches a value of 58¢ and is promptly reduced by the system feedbacks due to driver fuel, axial expansion, and sodium-coolant density changes. The important operating conditions in this type of transient are the temperatures at the peak point on the driver-fuel cladding. Note in Fig. 29 that, 107 sec after the initiation of the transient, the cladding temperature exceeds the uranium-stainless steel eutectic temperature. This long interval would provide more than sufficient time for administrative action. (Such action is not considered in this report.) The main point stressed in this analysis is that a fixed power-level trip would not protect against this component malfunction.

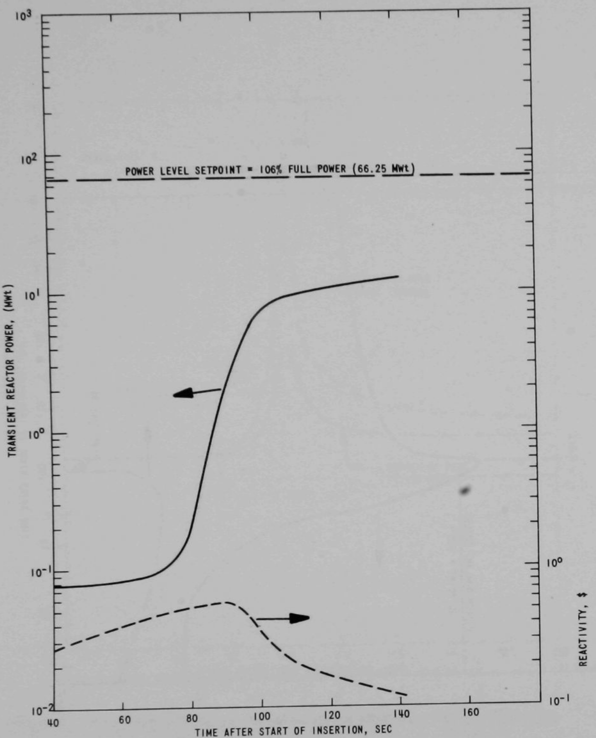


Fig. 28. Power and Reactivity Curves following Driving of EBR-II Safety Rods into Just-critical Core; Power-level-trip Setpoint at 66.25 MWt (106% of full power)

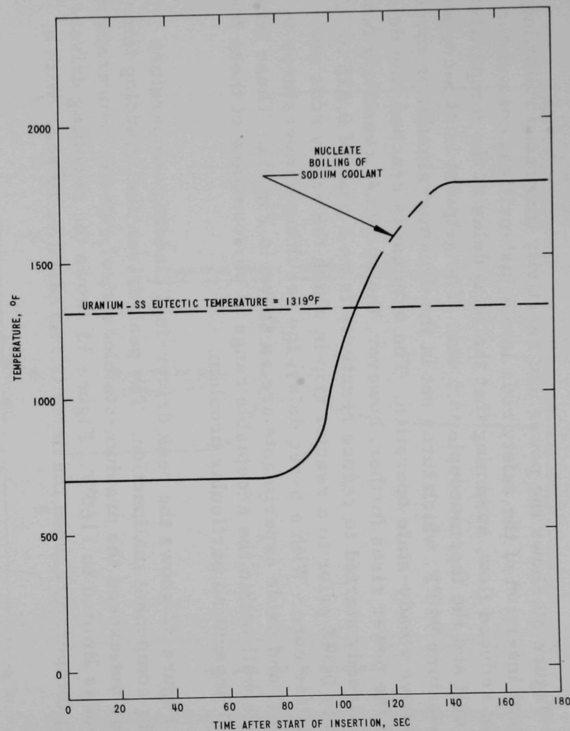


Fig. 29. Peak Driver-fuel Cladding Temperatures following Driving of EBR-II Safety Rods into Just-critical Core; Reduced Primary-coolant Flow; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

2. Trip from High Coolant-outlet Temperature

Figure 30 shows the power and reactivity increases following an inadvertent insertion of the safety rods into a just-critical core under conditions of reduced flow, assuming that the total delay of the temperature-sensing circuit and the thermocouple is 5 sec. The setpoint is at a coolant-outlet temperature 940°F, which turns out in the dynamic-simulation model to be 20°F above steady-state operation. The setpoint is reached, the delay occurs, and the power rises further; however, the system's reactivity feedbacks have already started to reduce reactivity from a peak of 0.57$ to approximately 0.34$ prior to a reactor trip in which the safety rods are driven out of the core. With a 5-sec delay, the resulting temperatures in the driver fuel and oxide experiments are as shown in Fig. 31. These temperatures are well within the acceptable range of performance of these fuel elements and present no particular problem.

Figure 32 shows the peak driver-fuel cladding temperature following this component malfunction. The peak driver-fuel cladding temperature does not exceed the uranium-stainless steel eutectic temperature of 1319°F, but is limited to 1160°F. Figure 33 shows the resulting driver-fuel cladding temperature when the delay is increased to 10 sec. With a

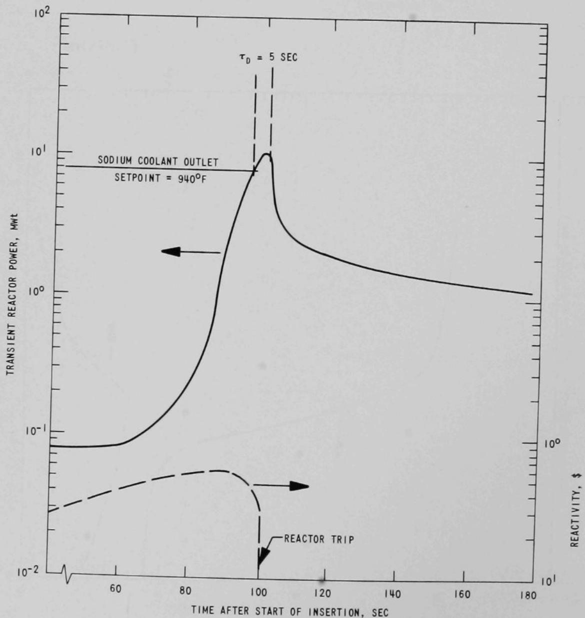


Fig. 30. Power and Reactivity Curves following Driving of EBR-II Safety Rods into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5-sec Delay

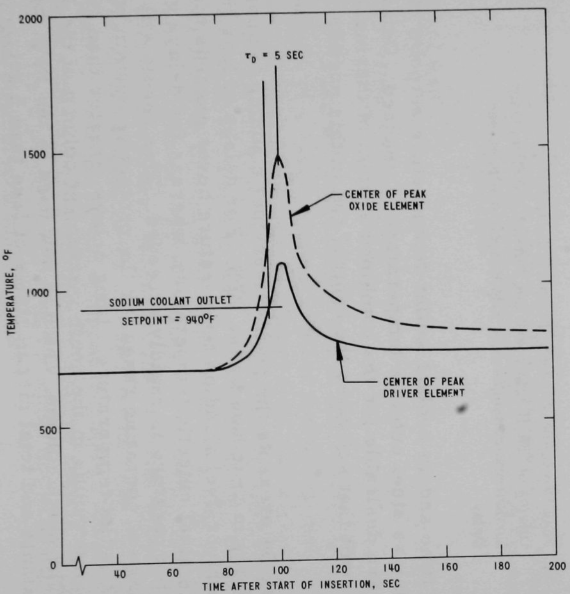


Fig. 31. Peak Fuel Temperatures following Driving of EBR-II Safety Rods into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5-sec Delay

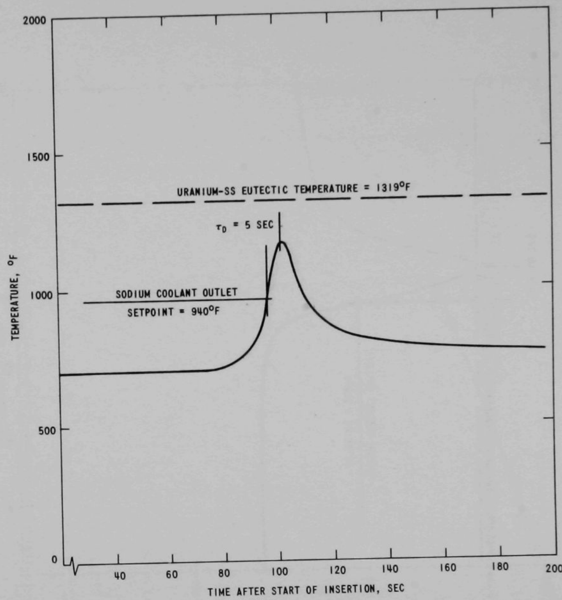


Fig. 32. Peak Driver-fuel Cladding Temperatures following Driving of EBR-II Safety Rods into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5-sec Delay

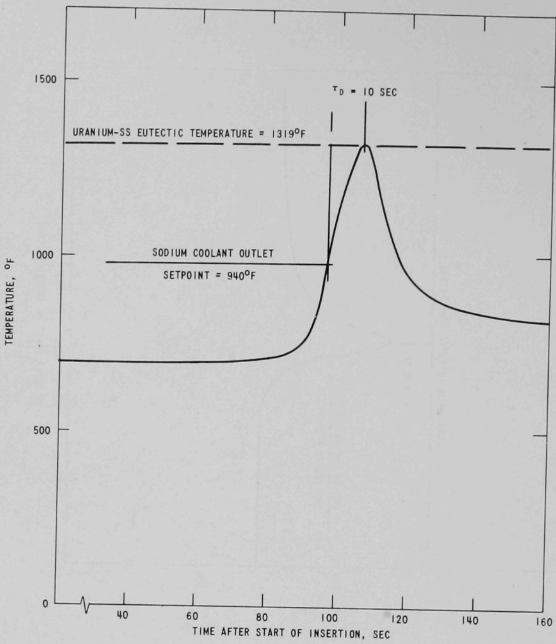


Fig. 33. Peak Driver-fuel Cladding Temperatures following Driving of EBR-II Safety Rods into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 10-sec Delay

10-sec response time and an coolant-outlet-temperature setpoint of 940°F, the uranium-stainless steel eutectic temperature is reached. Delays of less than 10 sec are desirable for the combined system of thermocouple and electronic circuit that activates the control rod in this portion of the PPS.

Figure 34 presents the peak driver-fuel cladding temperature as a function of delay in response by the PPS. For delays of less than 10 sec, the peak driver-fuel cladding temperature never exceeds the uranium-stainless steel eutectic temperature, whereas for delays greater than 10 sec, this temperature is rapidly exceeded. This result will be used in formulating the operating range for the protective system with respect to coolant-outlet temperature. Long delays, greater than 10 sec, are typical of thermocouples in the south instrument column of the EBR-II reactor vessel. The north column, which is now used in the PPS, has shorter time constants and total circuit times of less than 5 sec.

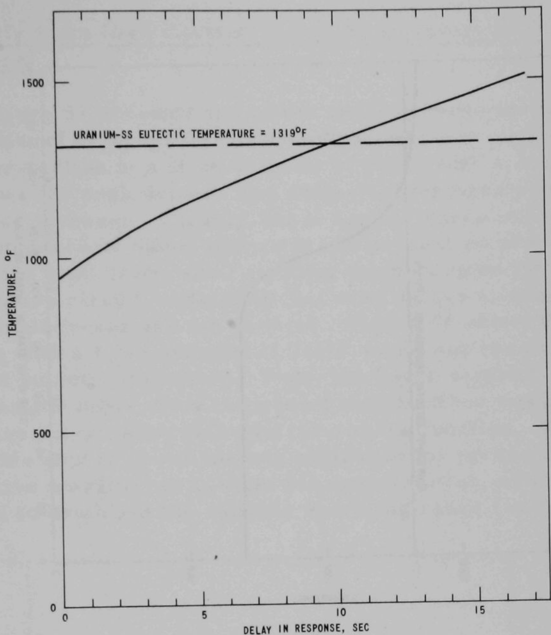


Fig. 34. Peak Driver-fuel Cladding Temperatures as a Function of Delay in Response of PPS to High Coolant-outlet Temperature, following Driving of EBR-II Safety Rods into Just-critical Core

B. Control Rod

1. Power-level Trip

Figure 35 depicts the power and reactivity increases following the insertion of a single EBR-II control rod into a just-critical core under conditions of reduced primary-coolant flow. It is assumed that no reactor-period trip is available and that the only protection available is a power-level trip set at 106% of full power (or 66.25 MWt). As the figure shows, the power level is limited by prompt temperature-induced reactivity feedbacks, and the system reactivity does not exceed 54¢ before being reduced by the system feedbacks. However, the uranium-stainless steel eutectic temperature is reached in the cladding of the peak driver-fuel element 140 sec after the start of the component malfunction (see Fig. 36). This is more than sufficient time for administrative control to have occurred; however, a fixed power-level setpoint would not have automatically protected against this control-rod malfunction.

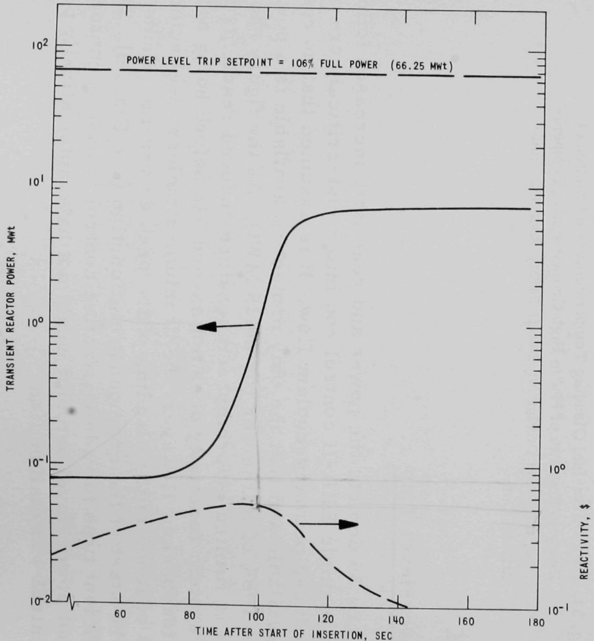


Fig. 35. Power and Reactivity Curves following Driving of Single EBR-II Control Rod into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

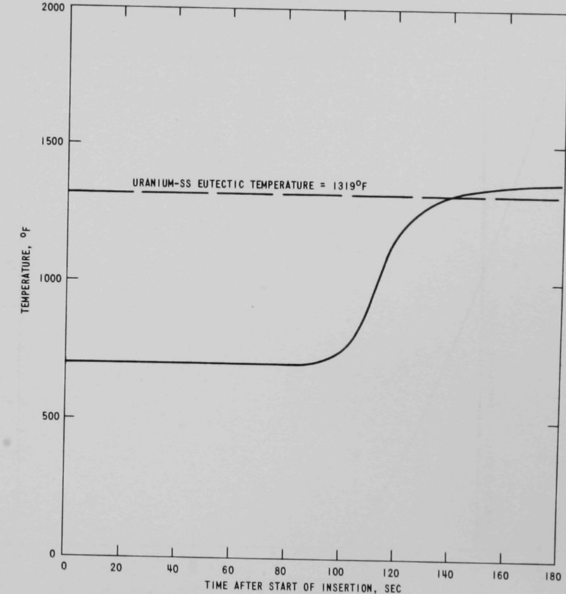


Fig. 36. Peak Driver-fuel Cladding Temperature following Driving of Single EBR-II Control Rod into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

2. Trip from High Coolant-outlet Temperature (initial just-critical core)

Figure 37 presents the power and reactivity increases resulting from the insertion of a single EBR-II control rod with only coolant-outlet-temperature protection at a fixed setpoint of 940°F with a delay of 5 sec. Figure 38 shows the peak driver- and oxide-fuel temperatures resulting from this power increase. Clearly, these temperatures are within the range of performance of these materials and present no problems. Figure 39 shows the peak driver-fuel cladding temperatures for various delays in the temperature circuit, indicating that even delays as long as 10 sec for this slow power increase are not critical. Figure 40 shows that delays as long as 15 sec with a fixed setpoint of 940°F would not result in uranium-stainless steel eutectic formation. Thus, the lower setpoints and shorter delays required for other cited component malfunctions would provide large protective margins for this control-rod malfunction. (Results to this point in this report do not include allowance for protective margins. These protective margins, to account for uncertainties, will be included in Section VIII to establish the suitable operating range for the PPS.)

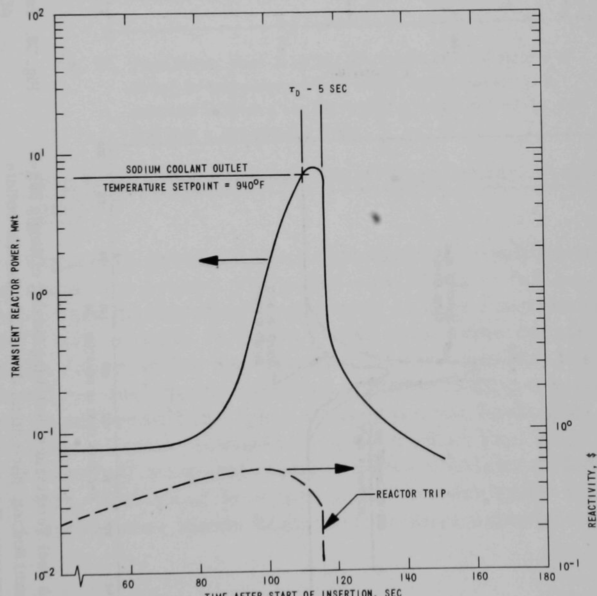


Fig. 37. Power and Reactivity Curves following Driving of Single EBR-II Control Rod into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5-sec Delay

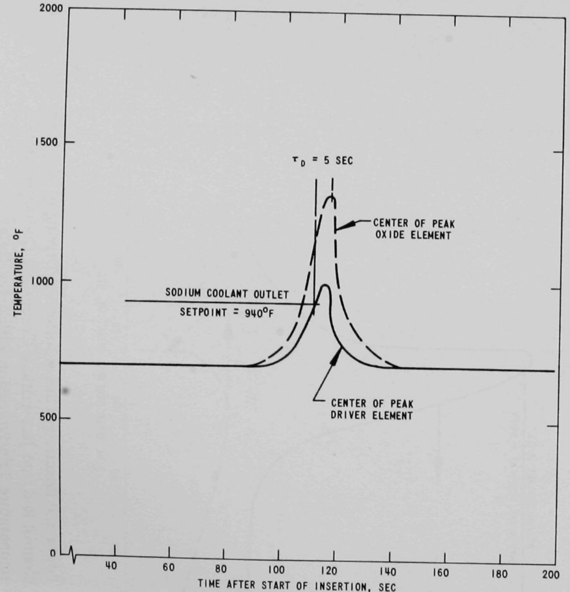


Fig. 38. Peak Fuel Temperatures following Driving of Single EBR-II Control Rod into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5-sec Delay

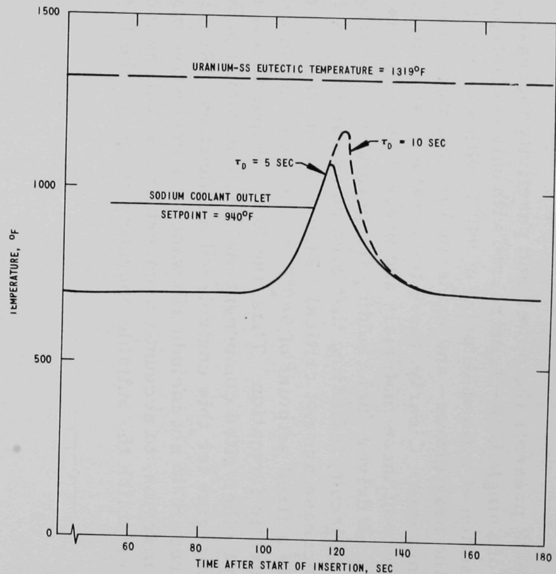


Fig. 39. Peak Driver-fuel Cladding Temperatures following Driving of Single EBR-II Control Rod into Just-critical Core; Only Coolant-outlet-temperature Trip Protection with 5- and 10-sec Delays

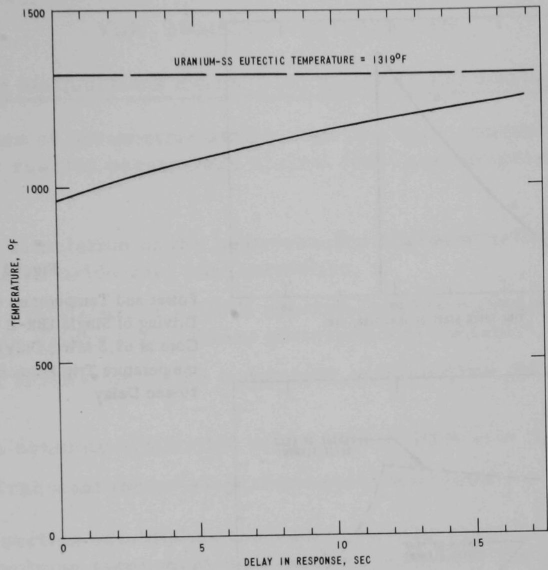


Fig. 40. Peak Driver-fuel Cladding Temperatures as Function of Delay in Response of PPS to High Coolant-outlet Temperature following the Driving of a Single EBR-II Control Rod into a Just-critical Core

3. Trip from High Coolant-outlet Temperature (initial full-power core)

Figure 41 presents the temperatures resulting from an inadvertent insertion of a control rod under conditions of full power and flow. Protection for this malfunction is assumed to be a fixed-temperature setpoint of 940°F with a delay of 10 sec. The peak oxide center temperatures are now in the range where damage may occur; however, the temperatures of the peak driver-fuel metal remain low under conditions of full flow. Longer delays will result in higher temperatures, leading to severe damage in the experimental oxide elements. Presented in Fig. 42 are the peak oxide and driver-fuel temperatures for various delays in the temperature-sensing circuits. Delays of less than 5 sec are desirable to prevent excessive temperatures inside high-performance experimental oxide elements.

C. Fuel Handling

Since the PPS is in the operate mode, control power to the fuel-handling equipment is not available; therefore, fuel-handling malfunctions are not considered here.

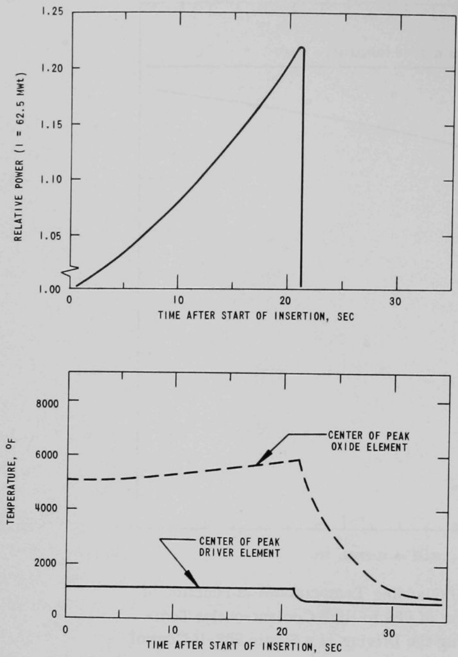


Fig. 41

Power and Temperature Curves following Driving of Single EBR-II Control Rod into Core at 62.5 MWt; Only Coolant-outlet-temperature Trip Protection at 940°F; 10-sec Delay

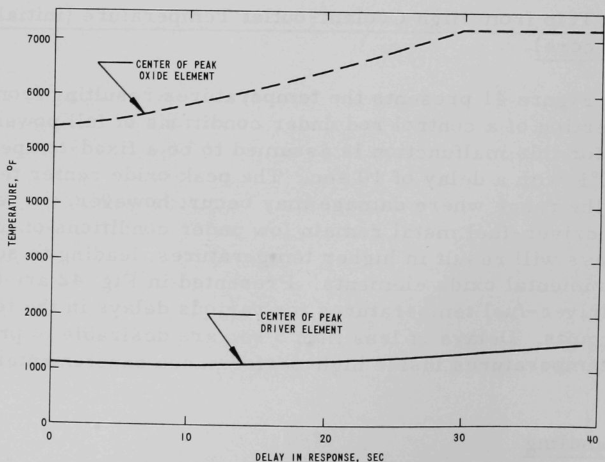


Fig. 42. Peak Fuel Temperatures as Function of Delay in Response of PPS to High Coolant-outlet Temperature (following the driving of single EBR-II control rod into core at 62.5 MWt)

VIII. SUMMARY OF RESULTS

A. Component Malfunctions during Fuel Handling and Startup

A series of parametric studies has now been completed. The principal dynamic reactor parameters of flux, flow, and temperature have been studied using

(1) A simulation of the heat-transfer characteristics of driver- and experimental-oxide-fuel subassemblies.

(2) A simulation of EBR-II reactor kinetics with prompt negative feedbacks from driver fuel, blanket material, and coolant.

(3) A simulation of the component malfunctions listed in the EBR-II HSR.

(4) A dynamic simulation of the EBR-II PPS with time delays.

(5) Transient material-performance restraints.

This section establishes the PPS operating range under dynamic conditions involving component malfunctions, with only the sensing of the reactor parameters of reactor period, power level, flow, or temperature. Many other PPS trips will protect against the malfunctions considered; these other trips have not been reviewed in this report. The qualitative justification for the protective margins considered in this section is presented in Appendix A.

1. Reactor-period Trip

Figure 43 shows the time required to reach a 25-sec reactor-period setpoint for various component malfunctions. A PPS operating range is identified in which period-trip protection would be available with present measured response times of the EBR-II PPS. There is an area above 1$/sec in which period-trip protection would not be available because of the long response times (as compared to transient times) that have been measured. Response times greater than 300 msec would not protect against a hypothetical malfunction involving the dropping of a central driver-fuel subassembly. However, for credible malfunctions under low-power conditions, the period trip would be the first level of protection and would occur early in the power increase, limiting temperatures to values near the coolant-inlet temperature, 700°F.

2. Power-level Trip

Figure 44 shows the transient reactor power at which eutectic temperatures would be reached under conditions of reduced flow. Indicated

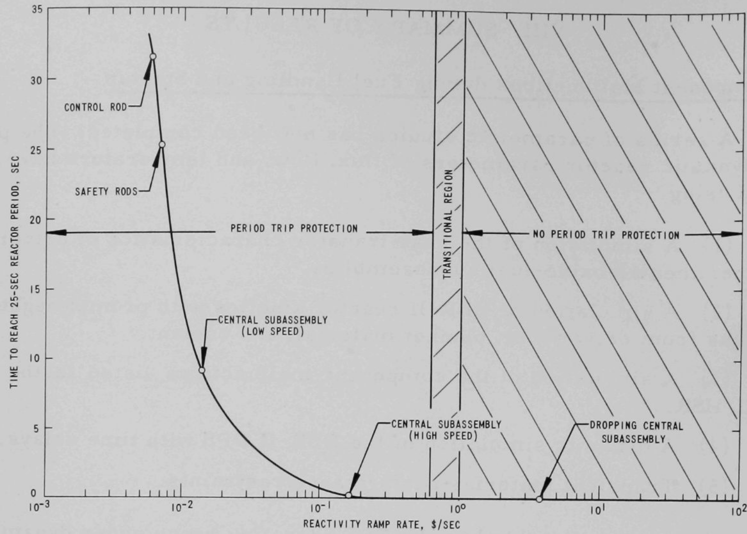


Fig. 43. Range of Effective Protection by Period Trip against Various Malfunctions (initial just-critical core)

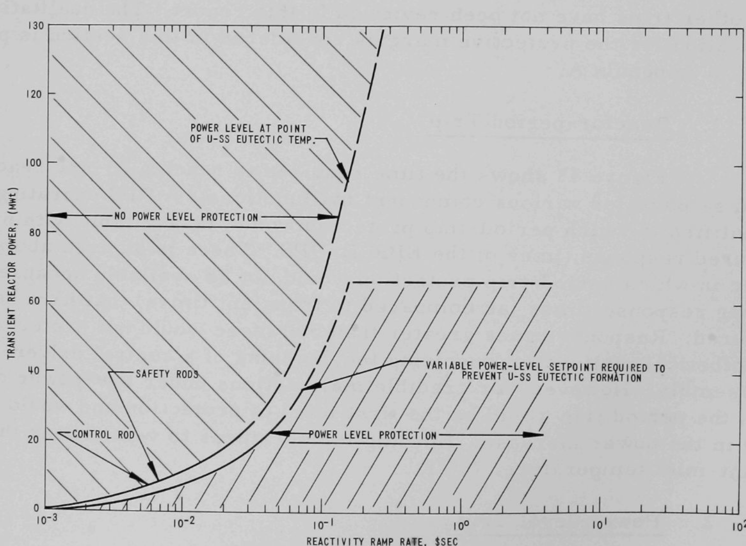


Fig. 44. Region of Power-level Protection That Would Prevent Formation of Uranium-Stainless Steel Eutectic with Various Malfunctions (initial just-critical core)

also is a PPS operating range of power-level protection if a variable power-level setpoint were available (e.g., in the common mode). A variable power-level setpoint would protect against all reactivity insertions resulting from the component malfunctions studied in this report (see Appendix C).

3. Temperature Trip

Figure 45 shows the peak driver-fuel cladding temperatures for a constant coolant-outlet-temperature setpoint with various response times, following two types of component malfunctions. The PPS operating range and the appropriate protective margin are indicated, and also the characteristics of the north- and south-column thermocouples and electronic circuits. As the figure shows, delays of less than 5 sec with a constant coolant-outlet-temperature setpoint of 940°F provide protection up to and including insertion of the safety rods.

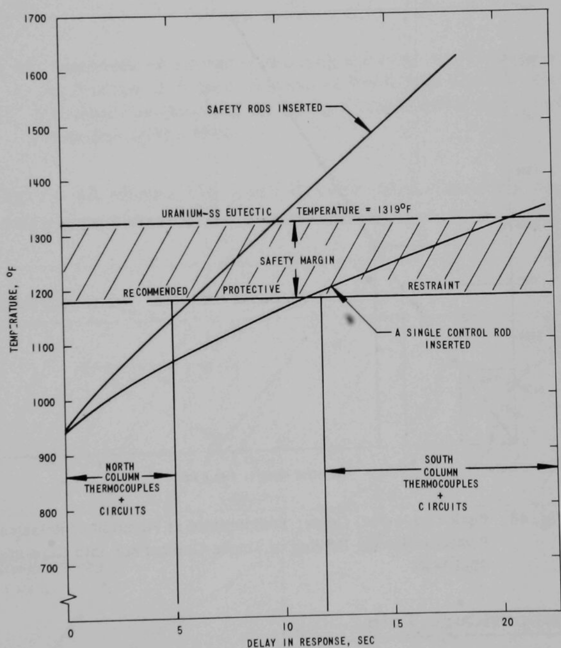


Fig. 45. Peak Driver-fuel Cladding Temperatures as Function of Delay in Response of PPS to High Coolant-outlet Temperature, following Various Malfunctions in Just-critical Core

B. Component Malfunctions during Full-power Operation

Two basic at-power malfunctions of components have been studied in this report:

(1) Insertion of a control rod.

(2) Loss of primary pumping power.

1. Power-level Trip

Figure 46 shows the PPS operating range in which peak oxide-fuel center temperatures are confined for various overpower conditions that follow the insertion of a single control rod. A power-level trip at 10% overpower easily maintains the oxide temperatures within postulated performance capabilities.

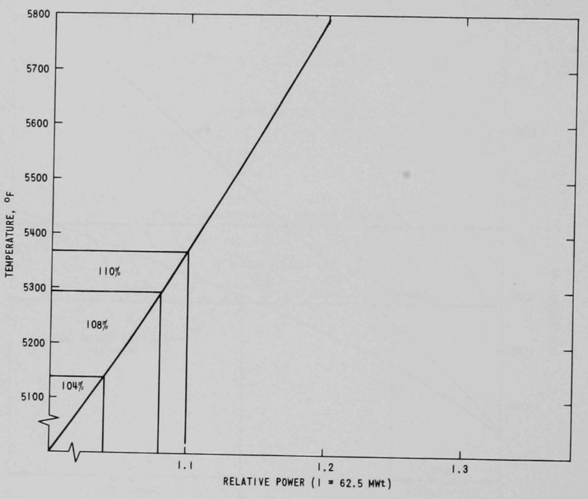


Fig. 46. Peak Oxide-fuel Center Temperature as Function of Relative Power, following Driving of Single Control Rod into Core at 62.5 MWt

2. Temperature Trip

Peak oxide temperatures resulting from the insertion of a single control rod at full power are presented in Fig. 47. The PPS operating range and the expected response times for the north and south thermocouple columns are indicated. Response times of less than 5 sec will provide an ample protective margin for the experimental oxide-fuel elements.

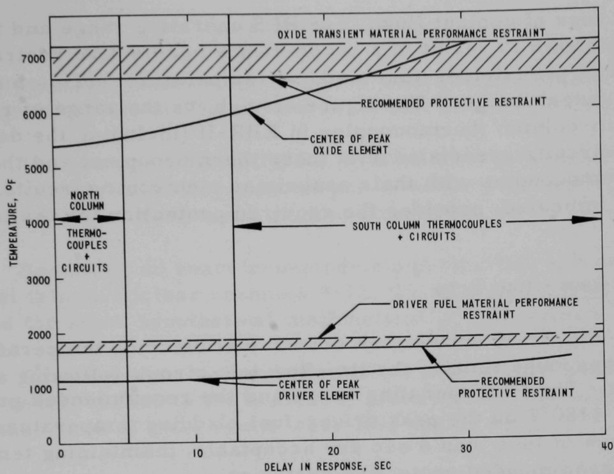


Fig. 47. Recommended Protective Restraints and Peak Fuel Temperatures as Function of Delay in Response of PPS to High Coolant-outlet Temperature (following the driving of single EBR-II control rod into core at 62.5 MWt)

Figure 48 shows the peak driver-fuel cladding temperatures as a function of response time of the high-coolant-outlet-temperature circuit,

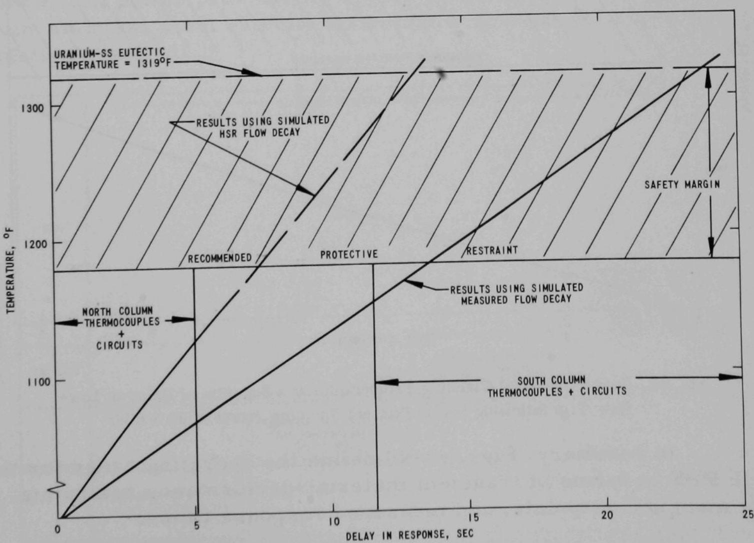


Fig. 48. Peak Driver-fuel Cladding Temperatures as Function of Delay in Response of PPS to High Coolant-outlet Temperature, following Loss of Coolant Flow at 62.5 MWt

following a loss of coolant flow. The PPS operating range and the protective margin are shown to result in a recommended protective restraint of 1180°F at the point of peak driver-fuel cladding temperature. (This includes all postulated uncertainties.) The figure also shows the range of response times for the south-column thermocouples in EBR-II (including the delay in the electronic circuits associated with these thermocouples) and the north-column thermocouples with their associated electronic circuits. The north column, as indicated, provides the required protection for the loss-of-flow transient.

3. Low-flow Trip

Figure 49 shows peak driver-fuel cladding temperatures as a function of response time in the low-flow trip circuit following a loss of coolant flow. The PPS operating range and the recommended protective restraint of 1180°F on the peak driver-fuel cladding temperature are indicated. Delays of less than 6 sec are acceptable, maintaining temperatures within the recommended protective restraint.

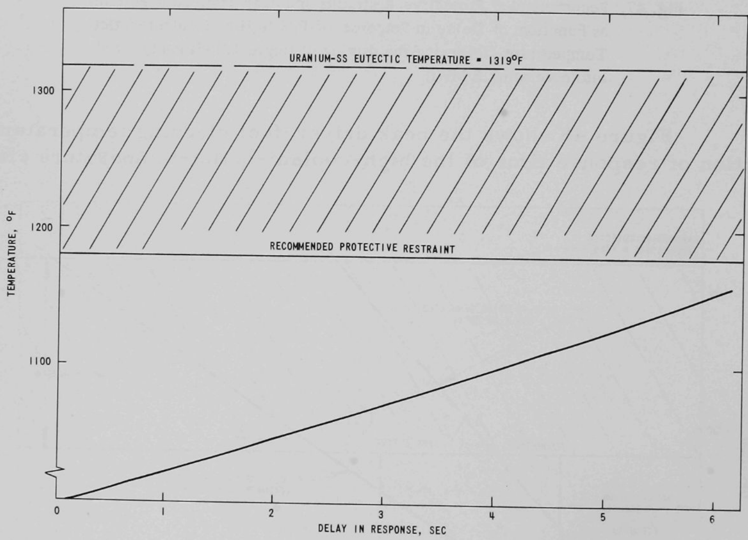


Fig. 49. Peak Driver-fuel Cladding Temperature as a Function of Delay in Low-flow Trip following Loss of Primary Pumping Power at 62.5 MWt

In summary, Figs. 43-49 define the operating range for the EBR-II PPS in terms of transient material-performance restraints, protective margins, setpoints, and measured response times.

IX. CONCLUSIONS

The principal conclusions obtained from these dynamic studies are presented qualitatively in Fig. 50 and are also listed below.

1. The present EBR-II PPS's with all shutdown modes operational will confine reactor operating parameters to within a desired operating range for all types of component malfunctions listed in the EBR-II HSR.

2. Assuming no reactor-period-trip protection, and only a high-power-level trip on nuclear channels 9-11, the peak fuel elements will be unprotected for some hypothetical malfunctions of components.

3. The trip from high subassembly-coolant-outlet temperature could be a useful backup to the nuclear channels in the operate mode, if the total response time of the circuits were less than 5 sec.

4. Assuming no power-level trips from nuclear channels 1, 2, 3, and 7, and assuming the existing PPS characteristics and only reactor-period trips from nuclear channels 4-6, protection is provided for all the component malfunctions considered, except the inadvertent dropping of a subassembly. (This last malfunction could lead to a prompt-critical reactor, since measured response times are too long to protect against this event.)

5. At 5.5% of full primary-system flow, a constant-power-level setpoint at full power provides no backup protection during startup, since uranium-stainless steel eutectic formation will occur for all component malfunctions studied.

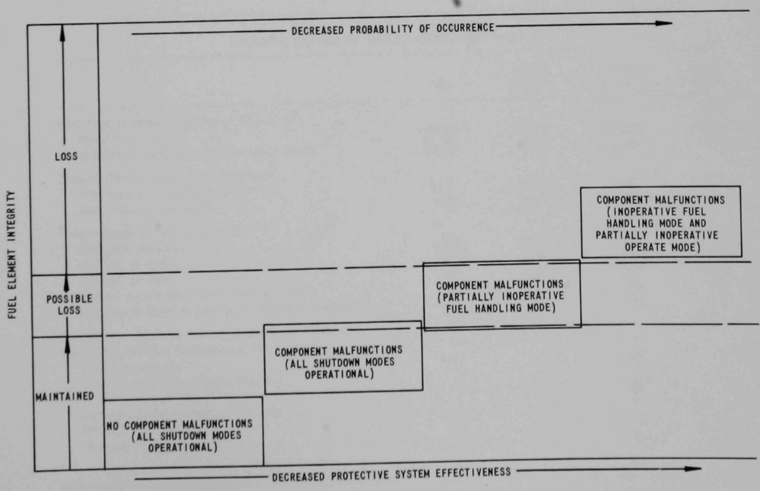


Fig. 50. Component Malfunctions and Levels of Protective-system Action

6. The present portions of the EBR-II PPS using the north-column thermocouples with delays of approximately 5 sec will provide adequate temperature backup for any of the hypothetical malfunctions considered during reactor startup.

7. The present low-flow setpoint and trip circuits are adequate for protecting against high driver-fuel-cladding temperatures with present measured response times.

APPENDIX A

Transient Material Limits and Protective Margins

Transient material limits were chosen to correspond the physical events that result in a change in a material's phase or form. These physical events were considered as "never-to-be-exceeded limits" and are listed below.

1. Uranium-stainless steel eutectic temperature = 1319°F.
2. Sodium coolant boiling = 1641°F.
3. Pressure or swelling rupture of cladding = 1800°F.
4. Melting of uranium-5 wt % fissium alloy = 1834°F.
5. Melting of experimental oxide fuel = 5074°F.
6. Vaporization of experimental oxide fuel = 7200°F.
7. Melting of experimental carbide fuel = 4800°F.

The most critical parameter in the study was the temperature of formation of uranium-stainless steel eutectic. The uncertainty analysis for 62.5-MWt operation, published in the HSR and its Addendum, was used to establish bands of temperature ranges associated with each critical material of the EBR-II driver fuel. Typical uncertainty analyses for experimental oxide fuel and cladding were used in a similar way to establish bands of temperature uncertainties for irradiation specimens. Results of these studies are presented in Tables VII-IX.

TABLE VII. Summary of Thermal Analyses for Maximum-temperature Region of Core and Blankets (Including uncertainty factors; reactor power = 62.5 MW)

	Core	Upper Blanket[a]	Inner Blanket	Outer Blanket
Heat Flux at Element Surface,[a] Btu/hr-ft[b]				
Maximum in zone	929,000	105,000	294,000	57,400
At point of maximum uranium temperature	571,000	84,800	279,000	300
Coolant Flow in Hottest Subassembly[a]				
Flow rate, gpm (sodium at 800°F)	93.7	93.7	31.0	5.9
Flow velocity, fps (avg)	15.8	14.2	19.3	4.2
Temperatures in Hottest Channel, °F				
Uranium, maximum	1,271	1,090	1,154	958
Coolant, at outlet	1,075	1,032	930	958
Coolant, at inlet	705	1,027	700	700
Coolant temperature rise, inlet to outlet	370	5	230	258
Coolant, at point of maximum uranium temperature	1,058	1,027	849	958
Temperature Rises in Hottest Channel at Point of Maximum Uranium Temperature, °F				
Through uranium	117	36	210	0
Through uranium-sodium interface	8	1	5	0
Through sodium "bond" layer	9	2	12	0
Through sodium-cladding interface	7	1	5	0
Through cladding	44	16	58	0
Through coolant film	28	7	15	0
Total element temperature difference	213	63	305	0

[a]Since these analyses were made, the uranium in the upper-blanket region has been replaced by stainless steel.
[b]No uncertainty factors included.
Note: "Uranium" here means fuel alloy or blanket uranium, as appropriate.

TABLE VIII. Uncertainty Analysis for Centerline Temperature of Typical Oxide Fuel

Thermophysical Property	Coolant ΔT = 162°F F	Coolant d	Film ΔT = 37°F F	Film d	Cladding ΔT = 81°F F	Cladding d	Gap ΔT = 698°F F	Gap d	Fuel ΔT = 2761°F F	Fuel d	Σd	$(\Sigma d)^2$
Neutron and gamma flux	0.07	11.34	0.10	3.70	0.10	8.10	0.10	69.80	0.10	276.10	369.04	136,191.00
Flow rate through subassembly	0.05	8.10									8.10	65.61
Flow profile within subassembly	0.05	8.10									8.10	65.61
Cladding dimensions					0.01	0.81					0.81	0.66
Gap dimensions							0.02	13.96			13.96	194.88
Fuel dimensions			0.01	0.37	0.01	0.81	0.01	6.98	0.01	27.61	35.77	1,279.49
^{235}U, ^{239}Pu concentrations	0.01	1.62	0.01	0.37	0.01	0.81	0.01	6.98	0.01	27.61	37.39	1,398.01
Cladding thermal conductivity					0.10	8.10					8.10	65.61
Gap conductance							0.30	209.40			209.40	43,848.40
Fuel thermal conductivity									0.15	414.15	414.15	171,520.00
Film H.T. coefficient			0.20	7.40							7.40	54.76
Power-level measurement	0.02	3.24	0.02	0.74	0.02	1.62	0.02	13.96	0.02	55.22	74.78	5,592.05
Transient overload	0.05	8.10	0.05	1.85	0.05	4.05	0.05	34.90	0.05	138.05	186.95	34,950.30

Note: F is the uncertainty factor; d is the temperature deviation, °F.

Total $\Sigma(\Sigma d)^2$ = 395,226

$\sqrt{395226}$ = 628.67°F

Nominal maximum fuel temperature (without uncertainty factors) = 4439°F
Increase in fuel temperature due to uncertainty factors = 629°F

Maximum fuel temperature with uncertainty factors = 5068°F

TABLE IX. Uncertainty Analysis for Mean Temperature of Oxide-fuel Cladding

Thermophysical Property	Coolant ΔT = 313°F F	Coolant d	Film ΔT = 26.4°F F	Film d	Mean Cladding ΔT = 24.6°F F	Mean Cladding d	Σd	$(\Sigma d)^2$
Neutron and gamma flux	0.07	21.89	0.10	2.65	0.10	2.48	27.02	730.08
Flow rate through subassembly	0.05	15.63					15.63	244.30
Flow profile within subassembly	0.05	15.63					15.63	244.30
Cladding dimensions					0.10	0.25	0.25	0.06
Fuel dimensions			0.01	0.27	0.01	0.25	0.52	0.27
^{235}U, ^{239}Pu concentrations	0.01	3.13	0.01	0.27	0.01	0.25	3.65	13.32
Cladding thermal conductivity					0.10	0.25	0.25	0.06
Film H.T. coefficient			0.20	5.31			5.31	28.20
Power-level measurement	0.02	6.25	0.02	0.53	0.02	0.50	7.28	53.00
Transient overload	0.05	15.63	0.05	1.33	0.05	1.24	18.20	331.24

Note: F is the uncertainty factor; d is the temperature deviation,°F.

Total $\Sigma(\Sigma d)^2$ = 1644.83

$\sqrt{1644.83}$ = 40.56°F

Nominal maximum mean-cladding temperature (without uncertainty factors) = 1064°F
Increase in mean-cladding temperature due to uncertainty factors = 41°F

Maximum mean-cladding temperature with uncertainty factors = 1105°F

Finally, it was assumed for the purposes of this report that these temperature bands of uncertainty represented one standard deviation (1σ). It was further assumed that the EBR-II protective margins should be based on at least 3σ. In subsequent studies, more definitive studies should be performed to add a firmer technical basis to these protective margins. Using the above conditions and qualifying assumptions, we obtained protective margins as listed in Table X.

TABLE X. Transient Material Performance Restraints and Protective Margins

Phenomenon	Damage Threshold Temp, °F	1σ	Protective Margin, 3σ	Protective Restraint
	Driver-fuel Elements			
Uranium-SS eutectic	1319	46	138	1181
Boiling of sodium coolant	1641	140[a]	520	1121
Melting of driver fuel	1834	58	174	1660
Cladding rupture	1800	140[a]	520	1280
	Oxide-fuel Elements			
Cladding rupture	1800	140[a]	520	1280
Oxide-fuel melting	5074	200	600	4474
Oxide-fuel vapor pressure	7200	200[b]	600	6600
	Carbide-fuel Elements			
Cladding rupture	1800	140[a]	520	1280
Carbide-fuel melting	4800	200	600	4200

[a]Allowance was made for locations of thermocouple sensors.
[b]Chosen on basis of early TREAT experiments.

APPENDIX B

Measurement of Response Time of the PPS Circuits

A comprehensive testing program was undertaken to measure the response times of those instrument channels in EBR-II considered to be of particular importance to the safety and control of the reactor. All measurements were made by scaling times off a chart which recorded the output from a Brush Clevite Model 260 six-channel recorder. Chart speed was always 125 mm/sec and was shown to be accurate to better than 0.2% in a preliminary test. In general, the time values cited below are average values for three measurements. Time did not permit all combinations of channels and tests.

1. Nuclear Channels

a. Startup Channels (1-3)

These channels record information from uranium-loaded fission chambers. The tests were carried out by disconnecting the regular fission chambers, which were left in place. The test chamber was placed in a paraffin barrel. Suitable beginning and final count rates were obtained by maneuvering two neutron sources in the barrel. A step was obtained by dropping the larger source. Time was measured from the appearance of neutron signals at the preamplifiers to the tripping of the control-power relay. Since these channels feed into a two-out-three circuit, one other channel had to be pretripped in order to get a measurement. The results are summarized in Table XI. Channels are designed to trip on both period and level signals. However, only one of the step inputs was in the range to operate the level trip. The nominal 1500-count/sec setpoint is actually somewhat higher.

TABLE XI. Response Times of Channels 1-3

Nature of Test Step, counts/sec	Time, in msec, to Period Trip			Time, in msec, to Level Trip		
	Channel 1	Channel 2	Channel 3	Channel 1	Channel 2	Channel 3
100-500	2020	2000	2100			
500-2000	1880	1980	1850			
1000-5000	1470	1640	1460	1470	730	600
2000-9000	1610	1570	1470			

Note: All three channels were set for a 25-sec period and a neutron level of 1500 counts/sec. In actuality, the level trip functioned only on the third step, as shown above.

b. Intermediate Channels (4-6)

These channels register the signals from compensated ion chambers, which were disconnected for these tests. The test signal consisted of a square step in current between the levels indicated in Table XII.

The signal was supplied by a signal generator specially built for this purpose and was introduced at the connector to which the chamber cable is normally connected. These channels trip on period, but not on level.

TABLE XII. Response Times of Channels 4-6

Nature of Test Input Step, amps	Time, in msec, to Operation of Control-power Relay		
	Channel 4	Channel 5	Channel 6
0.3×10^{-9} to 0.9×10^{-9}	310	-	-
0.3×10^{-7} to 0.9×10^{-7}	360	250	340
0.3×10^{-5} to 0.9×10^{-5}	250	250	340
0.3×10^{-4} to 0.9×10^{-4}	-	-	295
0.3×10^{-3} to 0.9×10^{-3}	260	245	-
10^{-9} to 10^{-6}	120	95	-
10^{-8} to 10^{-5}	-	-	112
10^{-7} to 10^{-4}	-	-	105
10^{-6} to 10^{-3}	100	125	-

Note: All three channels were set for a 25-sec period.

c. High-power Channels (9-11)

These channels use compensated ion chambers and were tested in the same way as the intermediate channels. However, in this case the trip results from high level rather than rate of change. Results are given in Table XIII.

TABLE XIII. Response Times of Channels 9-11

Nature of Test Input, amps	Time, in msec, to Operation of Control-power Relay			Time, in msec, to Release of Scram Clutch on No. 8 Control Rod		
	Channel 9	Channel 10	Channel 11	Channel 9	Channel 10	Channel 11
Step from 10^{-8} to 10^{-4}	32	29	32	26	23	26
Step from 10^{-6} to 10^{-4}	34	31	31	25	26	23
Step from 10^{-5} to 10^{-4}	36	31	29	26	25	24
Step from 0.3×10^{-4} to 10^{-4}	33	29	29	27	22	23

Note: For all tests the channels were set at their normal operating setpoints:

Channel 9 = 0.49×10^{-4} amps,
Channel 10 = 0.519×10^{-4} amps, and
Channel 11 = 0.44×10^{-4} amps.

2. Thermocouple Channels

These measurements were made by introducing a step voltage input corresponding to the temperatures shown in Table XIV. The times indicated apply only to the electronic channels and do not include the effects of the thermocouple time constants. Thermocouples 503V, 503Z, 503M, and 503Q measure individual subassembly-outlet temperatures. Thermocouple 507BK monitors mixed-mean reactor-outlet temperature.

TABLE XIV. Response Times of Thermocouple Electronic Channels, Exclusive of Thermocouple Time Constants

Simulated Temperature Step, °F, with Respect to Operating Setpoint	Time to Trip, msec				
	TC 503V	TC 503Z	TC 503M	TC 503Q	TC 507BK
-55 to +10	2230	-	-	-	-
-55 to +5	-	2680	2690	1950	1010
-45 to +5	2740	2710	2780	785	965
-35 to +5	2600	2190	220	1670	870
-25 to +5	2170	1900	1900	1220	1100
-15 to +5	1710	1480	1380	980	920

Note: Normal operating setpoints are: 503V = 905°F; 503Z = 870°F; 503M = 875°F; 503Q = 861°F; 507BK = 862°F.

3. Flow-sensing Channels

These channels consist of two types, those that measure flow and those that measure the rate of change of flow. Measurements were first made by cutting pump power at full flow and timing the interval from power off to tripping of the control-power relay. The simulated loss-of-flow measurements were made by introducing into the sensor channel a signal representing a sudden 10% loss of flow, time being measured from signal to tripping of the control-power relay. Results are given in Table XV.

TABLE XV. Response Times of Flow-sensing Channels

Test	Unit	Setpoint	Time, sec
Low flow	Pump No. 1	-6%	3.52
	Pump No. 2	-6%	3.78
	Total	-6%	3.75
Rate of change	Pump No. 1	3%/sec[a]	2.3
	Pump No. 2	3%/sec[a]	2.65
Rate of change with simulated loss of flow	Pump No. 1	3%/sec	1.15
	Pump No. 2	3%/sec	1.40

[a]Actual flow at the time of activation of these trips was about 97%.

APPENDIX C

Possible Backup Protection from a Variable High-power-level Trip on Nuclear Channels in Fuel-handling and Operate Modes

To demonstrate the merits of variable high-power-level protection in the common mode during fuel handling, the following assumptions are made:

(1) A gross loading error of ~1500 Ih has occurred, leading to a delayed-critical core.

(2) A complete breakdown in administrative control has occurred, and criticality is unnoticed.

(3) Level trips on channels 1-3 are inoperative, but all interlocks have been satisfied so that control power to the fuel-handling equipment is available.

(4) Period trips on channels 1-6 are inoperative.

(5) The only protection provided is a variable high-level trip on nuclear channels in the common mode.

1. Component Malfunction during Fuel Handling

a. Central Subassembly Driven into Core at Low Speed

Figure 51 shows what happens when a central subassembly is driven at low speed into a just-critical core under conditions of reduced flow, with protection from a setpoint at 25% of full power and a response time in the circuit of about 30 msec. The setpoint is reached and a reactor trip occurs. (For this malfunction, only the safety rods are driven out of the core.) Before the trip, prompt feedbacks have reduced the reactivity from the peak value of 68¢ down to 52¢.

Figure 52 shows that the peak temperatures in the fuel, now the only important temperatures, are well within the accepted range for driver-fuel and experimental subassemblies.

b. Central Subassembly Driven into Core at High Speed

Figure 53 shows the power and reactivity changes following the driving of a central driver-fuel subassembly at high speed into a delayed-critical core at reduced-primary-flow conditions. The only protection available in this simulation is a high-power-level trip. The power increase is terminated at 106% of full power, and the safety rods are driven out of the core. The reactor never achieves prompt criticality. The peak reactivity

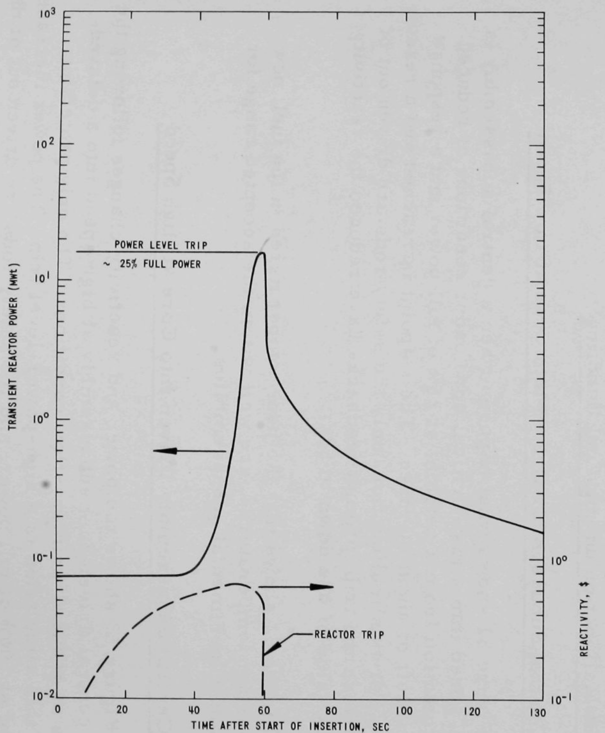


Fig. 51. Power and Reactivity Curves following Driving of a Central Driver-fuel Subassembly at Low Speed into Just-critical Core (reduced primary-coolant flow; power-level trip setpoint at 25% of full power)

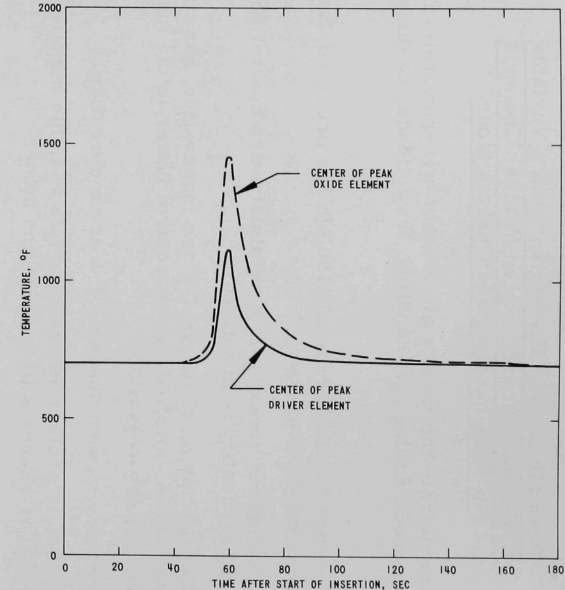


Fig. 52. Peak Fuel Temperatures following Driving of Central Driver-fuel Subassembly at Low Speed into Just-critical Core (reduced primary-coolant flow; power-level trip setpoint at 25% of full power)

is about 0.96$, and the prompt negative feedback turns around the transient before the removal of the safety rods. Figure 54 shows the peak driver-fuel and oxide-fuel temperatures following this power increase. The temperatures are well within the capability of the fuel, and no damage results.

c. Dropping of Subassembly

Figure 55 shows the result of inadvertently dropping a central driver-fuel subassembly into a just-critical core under conditions of reduced flow and with only a power-level trip to protect against this event. The trip point is passed, and the transient power rises to approximately 1400 MW for a fraction of a second. Safety rods are then driven out of the core, and the reactor power is safely reduced. For a short time the reactor is prompt critical, but the prompt negative feedbacks turn around the burst and the safety rods are driven out of the core. Figure 56 shows the resulting peak driver-fuel and oxide-fuel temperatures following this short power burst. As indicated, even at an extremely high setpoint of 106% of full power, this event is protected against. The power-level protection with its shorter response time (approximately 30-50 msec) protects against this maximum accident quite nicely, whereas a period trip with its longer response time (>300 msec) leads to an activity release from the reactor core in this hypothetical situation.

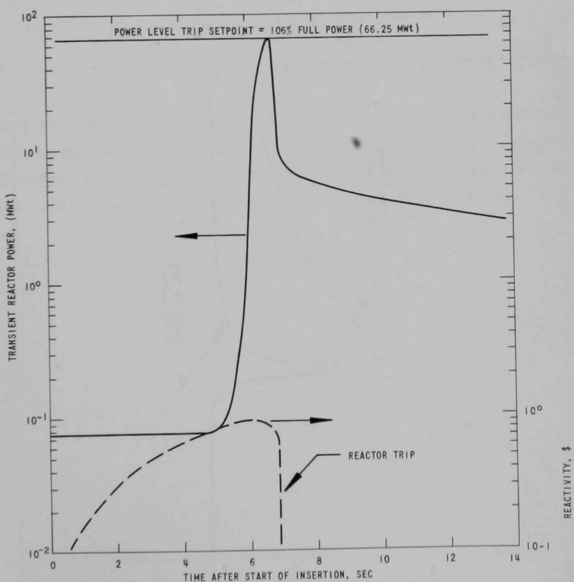


Fig. 53. Power and Reactivity Curves following Driving of Central Driver-fuel Subassembly at High Speed into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

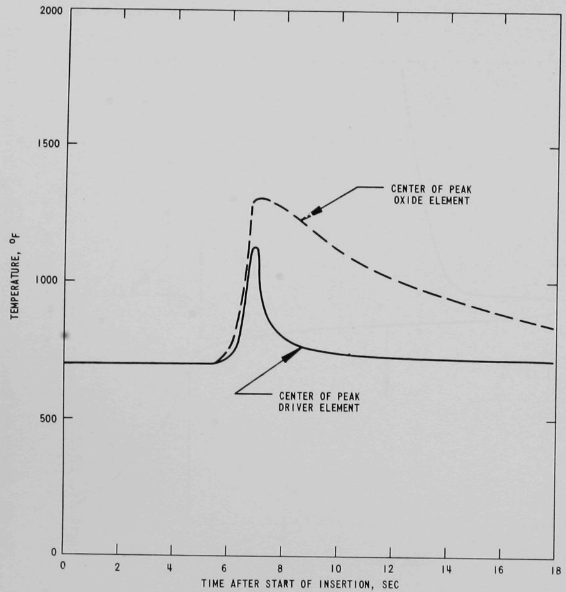


Fig. 54. Peak Fuel Temperatures following Driving of Central Driver-fuel Subassembly at High Speed into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

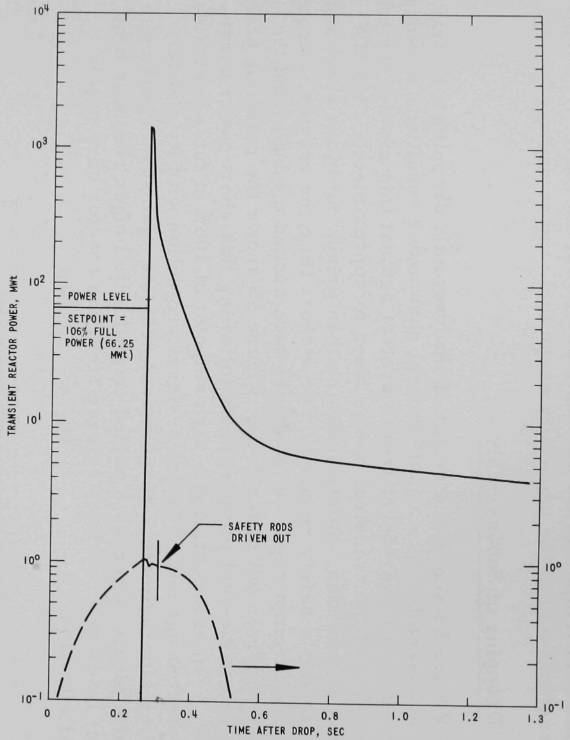


Fig. 55. Power and Reactivity Curves following Dropping of Central Driver-fuel Subassembly into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

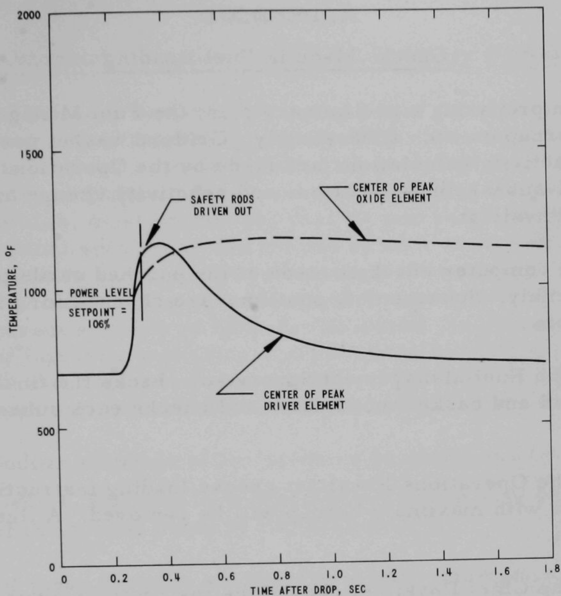


Fig. 56. Peak Fuel Temperatures following Dropping of Central Driver-fuel Subassembly into Just-critical Core; Power-level Trip Setpoint at 66.25 MWt (106% of full power)

Power-level protection based on a variable setpoint for nuclear channels in the common mode could adequately protect against all hypothetical malfunctions of components listed in the Hazard Summary Report. This is a useful conclusion for future PPS modifications.

APPENDIX D

Checks Made in Fuel Loading

1. In preparing a preliminary plan, the Fuel Management Supervisor checks the burnup on each subassembly. Grid and basket positions are also checked. Reactivity calculations are made by the Operations Physicist. Experimental-subassembly locations and reactivity change are checked by the Chief Physicist.

2. A computer check is made of the punched cards pertaining to each subassembly. Subassembly positions are checked for proper grid and basket locations.

3. The Fuel Management Supervisor checks the final loading plan for proper grid and basket coordinates and checks each subassembly for burnup.

4. The Operations Physicist checks loading instructions to see that subassemblies with maximum burnup will be removed. A final reactivity check is made.

5. The Chief Physicist checks the instructions again with respect to burnup and experimental-subassembly locations.

6. The Shift Supervisor checks the instructions again with respect to burnup and experimental-subassembly locations.

7. The loading instruction is approved by the Operations Manager.

8. When the loading is performed, grid and basket coordinates are visually checked. An inverse-count plot for each transfer is maintained in the control room. If after any transfer the inverse count has decreased to 50% of the original value, fuel handling is stopped. The loading sequence is planned so that reactivity-removal operations are performed early in the loading, and additions are performed later in the loading to keep the reactor as subcritical as possible during the major part of the loading.

9. Idaho Drafting checks the complete loading to determine that the correct subassembly has been removed from the grid. It prepares an updated grid-loading diagram.

10. A preliminary computer run-off is made to determine if all subassemblies reaching maximum burnup in previous runs have been removed.

11. An inverse-count plot is made during the approach to critical.

APPENDIX E

Abnormal Operations of EBR-II Primary Pumps

Loss of primary coolant flow in any power reactor is a serious occurrence requiring prompt action by the PPS. The usual loss-of-coolant-flow incident in a power reactor results in a coastdown of centrifugal coolant pumps to a minimum flow level maintained by backup or auxiliary pumps. In addition, most layouts for reactor-power-plant primary systems elevate the critical sections of the coolant system to provide a certain degree of convective flow. A sudden stoppage of centrifugal coolant pumps is considered an occurrence with an extremely low probability. In any event, the inherent stability of EBR-II will inhibit the consequences of a malfunction of this type, providing adequate time for PPS action. To demonstrate these inherent characteristics of the present EBR-II reactor and primary system, two additional loss-of-coolant-flow cases were studied.

1. Sudden stoppage of one primary coolant pump (no reactor trip).
2. Case 1, followed by a reactor trip on low flow and a coastdown of the second primary coolant pump.

The results of these dynamic studies are briefly summarized below.

The assumed characteristics of the fuel and blanket channels are those listed in Table II (p. 33) with the exception of the peak driver-fuel element. The power of this channel was increased from 9.832 to 11.000 Btu/sec to demonstrate the inherent stability of the EBR-II driver-fuel elements.

1. Sudden Stoppage of One Primary Coolant Pump with No Reactor Trip

Figure 57 shows the curves for reactor power and reactivity feedback following an inadvertent and sudden decrease in primary-coolant flow to 50% of full flow with no delay. The figure indicates that prompt reductions in power and reactivity follow this hypothetical decrease in flow. The basic reason for these prompt reductions is the momentary increase in material temperatures of the EBR-II metal driver-fuel elements, as shown in Fig. 58. Because of the very short time constant of these fuel elements (~0.3 sec) the temperature promptly increases, thereby introducing a large negative reactivity insertion of approximately -39 Ih. This reactivity insertion is introduced into the system in less than 2 sec, thereby terminating the event and limiting its consequences by means of the inherent stability characteristics of the EBR-II reactor. The prompt feedback of the driver-fuel elements continues to decrease the power of the reactor past this initial interval of 2 sec.

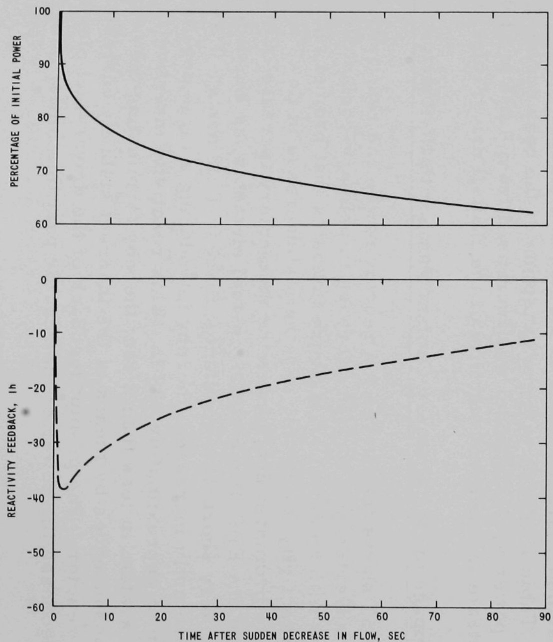


Fig. 57. Power and Reactivity Feedback following Inadvertent and Sudden Decrease in EBR-II Primary-coolant Flow (no reactor trip)

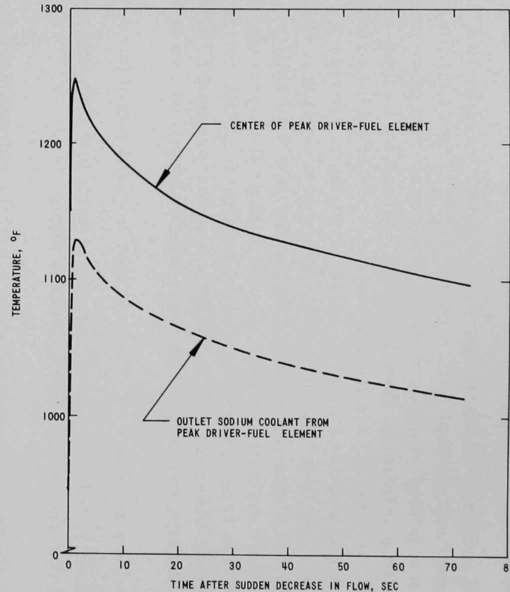


Fig. 58. Material Temperatures in Peak Driver-fuel Element following Inadvertent and Sudden Decrease in EBR-II Primary-coolant Flow (no reactor trip)

Figure 59 shows the profiles of driver-fuel temperatures along the centerline radius of the fuel pin. (Each of the three nodes represents equal volume regions in the fuel pin.) Note that the metallic driver fuel responds to the sudden flow reduction by an increase in temperature, and this response is followed by a decrease in temperature due to the reduction in reactor power. The experimental oxide fuel elements only respond to the reduction in power, as shown in Fig. 60. The slower response of the oxide fuel elements is directly traceable to the inherent physical properties of oxide fuel and its associated time constant of ~4 sec. The surface of the oxide fuel does exhibit a small response to the initial loss of flow, as depicted in Fig. 60; however, the principal response is to the reduction in reactor power.

2. Sudden Stoppage of One Primary Coolant Pump, Followed by a Reactor Trip on Low Flow and a Subsequent Coastdown of the Second Coolant Pump

As noted in Section 1 above, the inherent properties of the EBR-II driver-fuel elements and the associated temperature-induced reactivity-feedback networks control the conditions in the reactor following a reduction in coolant flow to 50% of full flow. A reactor trip due to low flow would occur 1-2 sec after sudden stoppage of one primary coolant pump. A subsequent coastdown of the second coolant pump would be an extremely unlikely occurrence, but is studied to demonstrate some of the engineering features of the EBR-II primary-coolant system.

Figure 61 shows the power variation following:

a. A sudden stoppage of one primary coolant pump, and reduction of coolant flow to 50% of full flow.

b. An assumed reactor trip at 1.0 sec on low flow.

c. A coastdown of the second coolant pump.

Notice that the power has been reduced before the reactor trip. Clearly, the first part of this incident (less than 1 sec) is controlled by the inherent properties of the reactor.

Figure 62 shows the driver-fuel surface and cladding temperatures following this occurrence. In this study, the auxiliary and convective coolant flow was assumed to be greatly reduced to 1.65% of full flow from its normal value of 6% of full flow.

With a reactor trip, the fission power is reduced to very low levels, leaving only the afterglow fission-product heat content. The peak in the

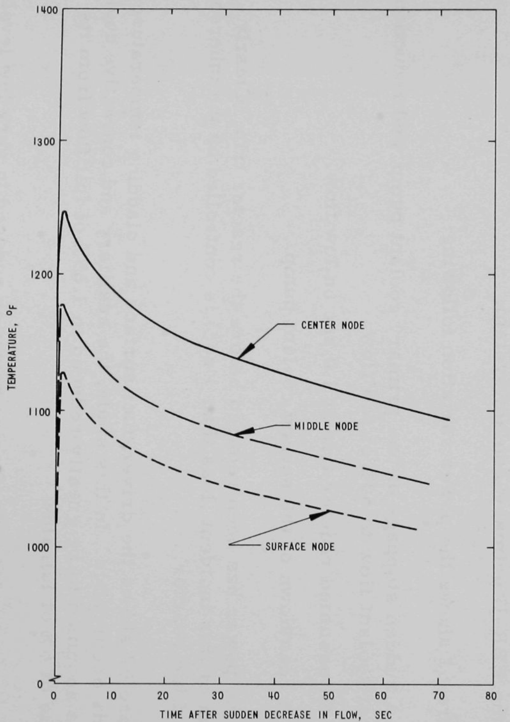


Fig. 59. Radial Temperature Variations in Peak Driver-fuel Element following Inadvertent and Sudden Decrease in EBR-II Primary-coolant Flow to 50% of Full Flow (no reactor trip)

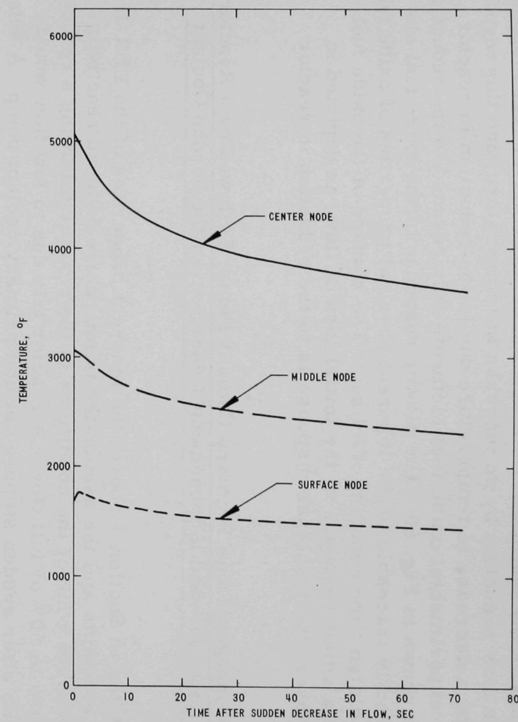


Fig. 60. Radial Temperature Variations in Peak Oxide-fuel Element following Inadvertent and Sudden Decrease in EBR-II Primary-coolant Flow to 50% of Full Flow (no reactor trip)

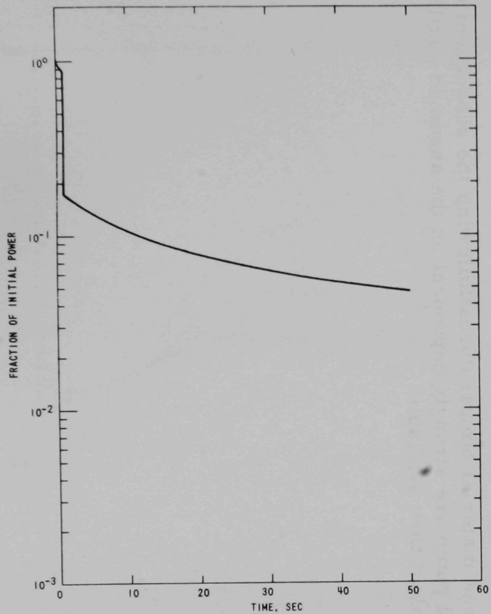


Fig. 61. Power Variation following Sudden Stoppage of Primary Pump No. 1, Peaking of Peak Driver-fuel Temperatures, and Then Reactor Trip and Coastdown of Pump No. 2 (reactor trip at t_0 + 1.0 sec; auxiliary and convective flow = 1.65% full flow)

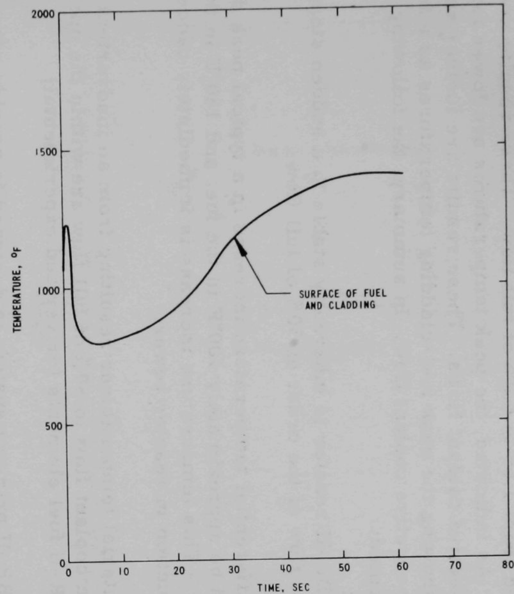


Fig. 62. Material Temperatures on Peak Driver-fuel Element following the Sudden Stoppage of Primary Pump No. 1, a Peaking of Peak Driver-fuel Temperatures, and Then a Reactor Trip and Coastdown of Pump No. 2 (reactor trip at t_0 + 1.0 sec; auxiliary and convective flow = 1.65% full flow)

material temperatures, shown in Fig. 62, is due to a short interval of time when the available coolant flow is less than required to remove the afterglow heat content. To demonstrate this factor, the auxiliary and convective coolant flow was varied from 1.65 to 6.0% of full flow. These results are presented in Fig. 63. As indicated, the peak temperatures are lower for higher auxiliary and convective coolant flows. These results are further summarized in Fig. 64, showing the peak fuel-cladding temperatures as a function of auxiliary and convective coolant flow. In summary, the following conclusions are presented:

a. The EBR-II reactor is inherently stable to a sudden stoppage of primary-coolant flow of the order of 50% of full flow.

b. The differential temperature increase in a typical peak driver-fuel element would be approximately 100°F in the fuel and 180°F in the associated coolant. This temperature increase is immediately canceled by the prompt reduction in reactor power.

c. All material temperatures resulting from an inadvertent and sudden decrease in coolant flow to 50% of full flow are within the operating capabilities of EBR-II fuel elements (driver and experimental).

d. The EBR-II primary system is designed to provide adequate auxiliary and convective coolant flow following a sudden stoppage of one pump, followed by a reactor trip and coastdown of the second coolant pump.

e. The peak material temperatures following the subsequent coastdown of the second pump are strongly dependent on the assumed auxiliary and convective coolant flow available.

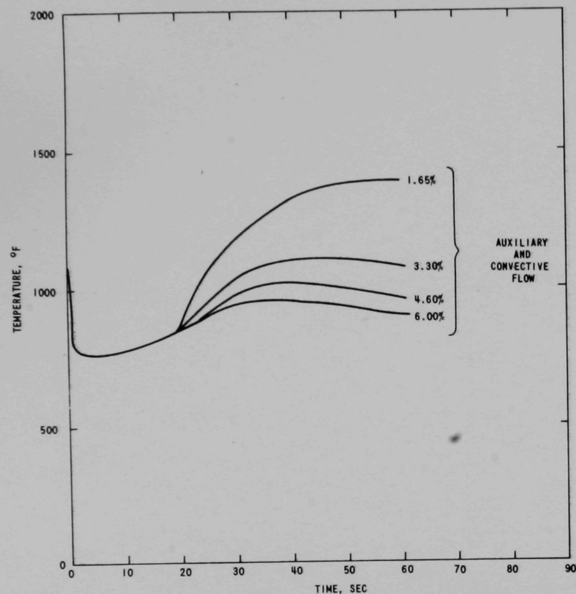


Fig. 63. Peak Driver-fuel-cladding Temperatures following Sudden Stoppage of Primary Pump No. 1 and Coastdown of Pump No. 2 (reactor trip on low flow [94% full flow] at t_0 + 0.050 sec)

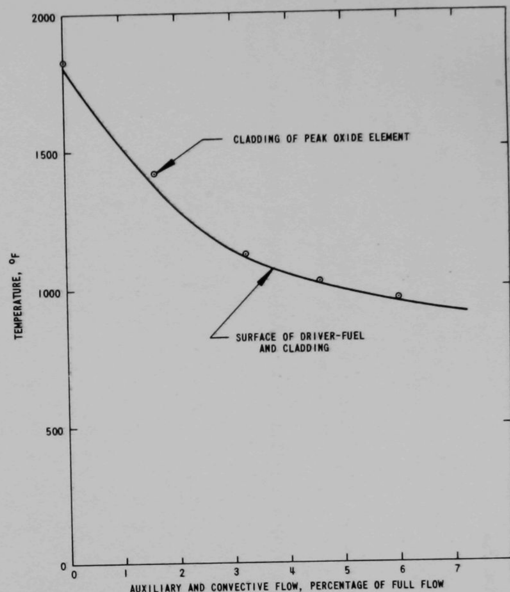


Fig. 64. Peak Fuel-cladding Temperatures following Sudden Stoppage of Primary Pump No. 1 and Coastdown of Pump No. 2, vs Auxiliary and Convective Flow

ACKNOWLEDGMENT

I wish to thank G. S. Brunson, J. R. Karvinen, and R. A. Call for providing Appendix B.

REFERENCES

1. L. J. Koch *et al.*, *Hazard Summary Report: Experimental Breeder Reactor II (EBR-II)*, ANL-5719 (May 1957).

2. L. J. Koch, W. B. Loewenstein, and H. O. Monson, *Addendum to Hazard Summary Report: Experimental Breeder Reactor-II (EBR-II)*, ANL-5719 (Addendum)(June 1962).

3. R. A. Blaine and R. F. Berland, *Simulation of Reactor Dynamics, Volume 1, A Description of AIROS-IIA*, NAA-SR-12452 (Sept. 5, 1967).

4. *Reactor Development Program Progress Report, October 1968*, ANL-7513, pp. 46-47.